

RELIABILITY STUDIES IN SPENT FUEL STORAGE AND TRANSPORTATION

By
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NUCLEAR ENGINEERING AND TECHNOLOGY PROGRAM

INDIAN INSTITUTE OF TECHNOLOGY, KANPUR

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RELIABILITY STUDIES IN SPENT FUEL STORAGE AND TRANSPORTATION

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DEEPAK CHATTERJEE

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CERTIFICATE

This is to certify that the work presented in this thesis entitled, 'Reliability Studies in Spent Fuel Storage and Transportation', by Deepak Chatterjee, has been carried out under my supervision and has not been submitted elsewhere for the award of degree.

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ABSTRACT

A model has been developed to estimate the risk to people during storage of spent fuel at the reactor site and during transportation of the spent fuel bundles to the reprocessing plant. The model begins with the calculation of fission product inventory in a typical CANDU reactor. The fission products decay with time, and the production of their daughters has a marked effect on the decaying activity of the total inventory.

Fault trees are developed to compute the accident probability in both the Spent Fuel Storage Pool and in the transportation of the spent fuel bundles to the reprocessing plant. The exposure to people, both due to beta and gamma rays following the dispersion of the radioactive plume under different weather conditions has been estimated. The risk to people through external exposure is presented as the result.

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Deepak Chatterjee

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CHAPTER I

INTRODUCTION

Nuclear power plants and their fuel-reprocessing facilities generate various types of radioactive wastes and the safe disposal of these wastes is an integral part of the use of nuclear energy. Radioactive waste handling, from nuclear power plant to final dumping involves several stages and many personnel, as shown in Fig. 1.1. These stages includes cooling of the spent fuel in the Spent Fuel Storage Pool (SFSP), transportation to the reprocessing facility, transportation to and final dumping at the dumping site.

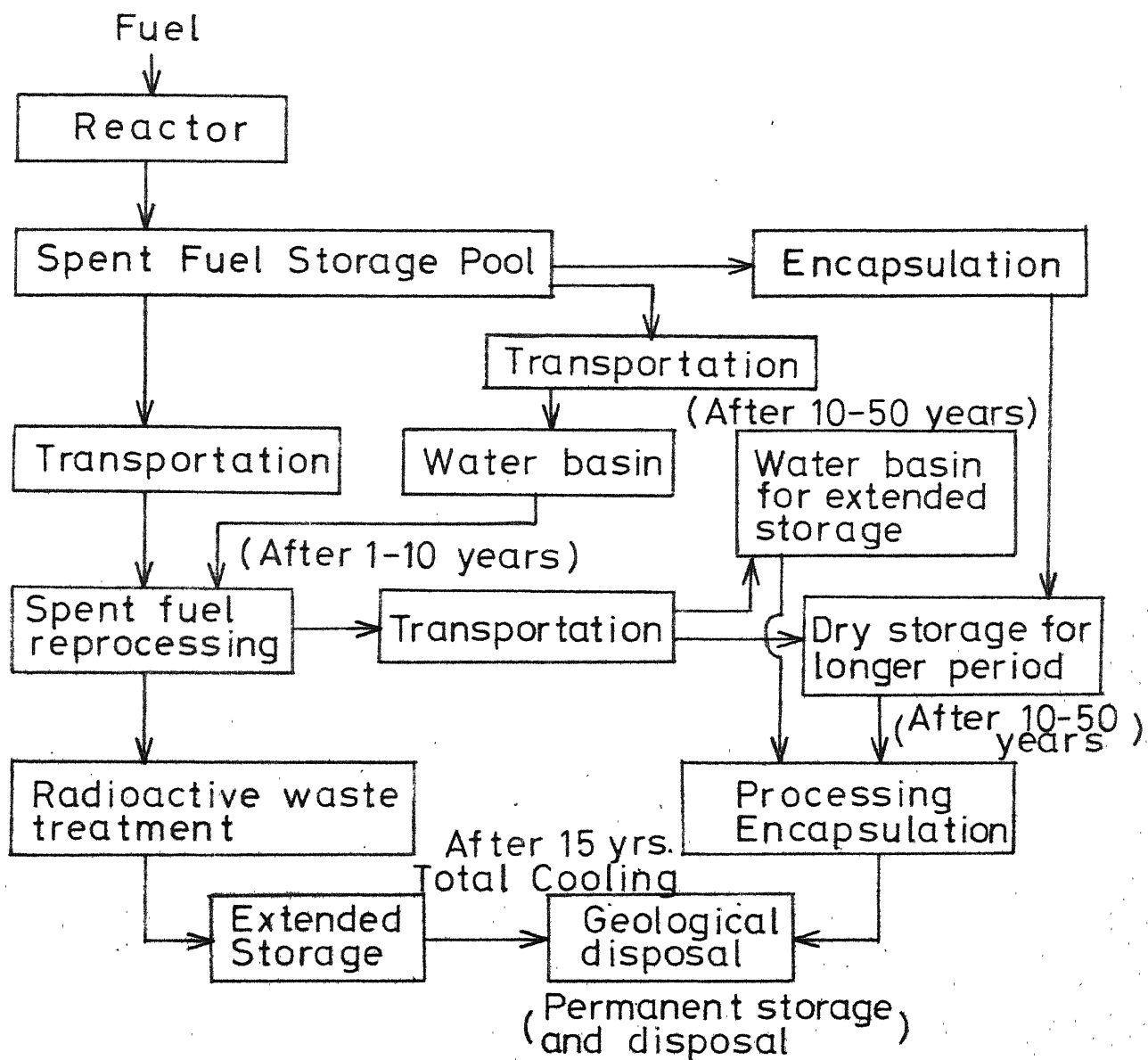
India has a 200 MWe CANDU unit operating and another of same capacity awaiting commissioning at Kota, Rajasthan. Two units of 200 MWe each are awaiting commissioning at Madras, two units of 230 MWe under construction at Narora, and another similar pair of units proposed somewhere in Gujarat. It becomes obvious that large amount of activity will have to be handled in future. Presently the fuel reprocessing is done at Tarapur and there are plans to build few more of them near different reactor sites.

Fuel reprocessing for CANDU reactor spent fuel is not undertaken in European countries and Canada as it is

found uneconomic. These countries have access to uranium enrichment facilities for their needs to run either light water reactors or breeder programs. However, the case is different for India. India requires plutonium as a substitute to enriched uranium for the subsequent fast breeder reactor phase of our nuclear power program. Plutonium can be obtained from natural uranium fuelled thermal reactors. As shown later, in Section 2.2, fissile plutonium is produced along with the fission products during the reactor operation in fuel bundles. This plutonium can be recovered by chemically reprocessing the spent fuel. Thereafter, as shown in Fig. 1.1, the depleted fuel with fission products is stored for an extended period, and finally disposed off in geological formations. Any accident or faulty handling at any of the stages involved, as shown in Fig. 1.1, poses a problem of radioactive release to environment and exposure to workers. There is some existing opinion that it is necessary to define an acceptable level of safety or in the alternative, an acceptable level of risk. This is amply brought forward by the IEEE special issue on Nuclear System Reliability and Safety [1].

1.1 CONCEPT OF RISK:

An acceptable risk to the society depends on several factors like time, nature and affluence of society, degree of technological progress, beliefs and literacy. To assess



Spent Fuel History Model

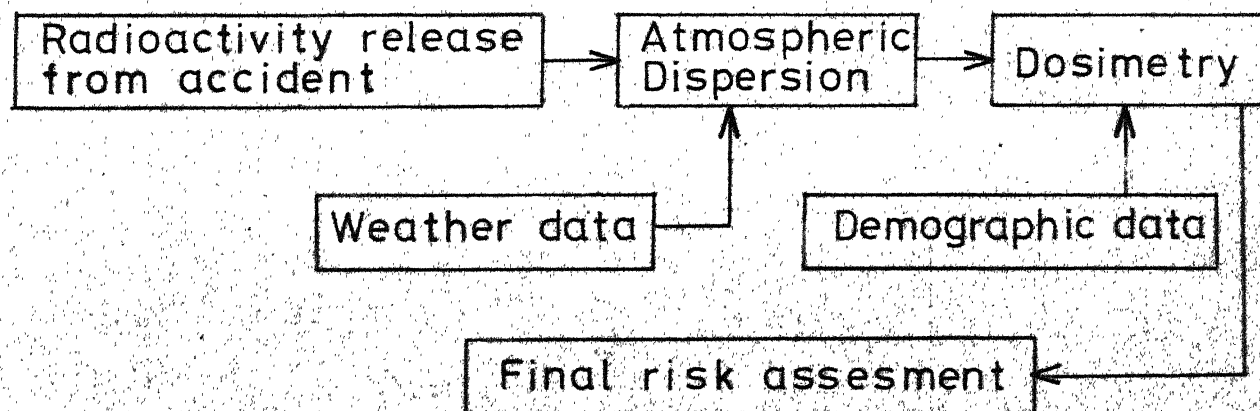


Fig.1-1 Schematic Consequence Model

the societal risk is therefore to confront a very difficult problem. According to Otway et al [2], 'The measurement of social values and their reconciliation with technical risk estimates through the framework of formal decision-making methodologies is defined as risk evaluation.' Therefore the best method seemingly possible is to compare the risk posed by a new technology with the risk posed by an already accepted technology.

To bring clarity to the term 'risk', Reactor Safety Study [3] defines risk to encompass potential fatalities and injuries to people and property. Societal risk from accidents is defined as

$$\text{RISK} \left[\frac{\text{Consequence}}{\text{Time}} \right] = \text{Frequency} \left[\frac{\text{Events}}{\text{Unit time}} \right] \times \\ \times \text{Magnitude} \left[\frac{\text{Consequences}}{\text{Events}} \right]$$

Another definition given by Kastenberget al. [4] is

$$\text{RISK} = [\text{Probability of occurrence of event}] \times \\ \times [\text{Consequence per event}] \quad (1.1)$$

This definition in more detailed form is forwarded by Smith et al [5] as,

$$\text{RISK} = A \times B \times C \times D \times E \quad (1.2)$$

where,

- A = probability of release sequence,
- B = release quantity
- C = relevant measure(s) of the radioactive material characteristics (isotopic composition, size distribution, solubility, volatility, etc.)
- D = a measure of environmental transport path efficiency
- E = a measure of population distribution characteristics and habits

The essence of the three definitions is however the two terms viz. probability and consequence of a particular event. However, the consequence assessment is the toughest problem since it usually includes injuries (fatal or non-fatal) to people and damage to property. The approaches prior to Reactor Safety Study handled the problem by converting the personnel injuries to lost man hours and property damage to monetary penalties. Reactor Safety Study [3] however, selected the following four types of consequences,

- a. Early fatalities
- b. Early illness
- c. Late health affects attributable to the accident
- d. Property damage.

1.2 RISK ASSESSMENT:

To estimate 'risk' in its totality, we require information of geographical, geological and demographic details, accident data at all stages and biological consequences of exposure to radiation. To compute risk it is also necessary to know the probability of radioactive material leakage following an accident. Unfortunately, for the reactors and all its associated stages shown in Fig. 1.1, either due to complication involved, or due to use of advance engineering systems, with inherent high reliability, coupled with precautions taken to avoid any accident due to public awareness and outcry, there is a serious lack of data which makes conventional analysis impossible. Therefore the Event Tree and Fault Tree methodology are very useful for risk assessment. Reactor Safety Study [6] uses the above methodology to analyze numerous accident chains. For each accident chain, a probability of occurrence is computed and the resulting consequence is evaluated. The collection of probabilities and consequences for the various accident chains gave the required point estimates, from which risk calculations can be performed.

1.3 FAULT TREES:

Fault tree analysis was introduced by Bell Telephone Laboratories in 1961 for performing safety evaluations of

launch control systems for the Minuteman Program [7], Fault tree analysis uses BOOLEAN ALGEBRA to depict component failures, and through AND and OR combinatorial logic produce system failure. Fault tree analysis is a deductive method whereby the analyst starts with the undesired state of the system as defined by the TOP event, and then proceeds to examine the system, component by component, for these combinations of component fault states that result in the undesired system state.

The relationship of these events to the TOP event are described by the fault tree equation by using Boolean operators i.e., effectively adding probabilities on 'OR' gates and multiplying probabilities on 'AND' gates. These events are then individually treated as TOP events and the process continued till we reach events for which data is available. Care has to be taken to avoid omissions, repetitions and logical discrepancies.

Some of the advantages of Fault tree analysis are [8]:

- (1) Directing the analyst to ferret out failures in a deductive way.
- (2) Providing a graphical aid giving system management visibility to those removed from the system design.
- (3) Providing options for qualitative or quantitative system reliability analysis.

- (4) Allowing the analyst to concentrate on the particular system at a time.
- (5) Providing the analyst with genuine insight into system behaviour.

The drawbacks listed are

- (1) Its inability to model failures that cannot be decomposed through binary logic e.g. system dynamic failure.
- (2) High cost involved and high time consumption,
- (3) Possibility of logical discrepancy in the fault tree, and repetition if constructed manually.

For a small tree, fault tree construction may look attractive. However, for a big multilevel fault tree, the process of finding a value or an expression for the resultant probability can be cumbersome. Several computer codes are available to analyze fault tree quantitatively. In many of these codes the numeric probabilities of failure are computed by means of Vesely's Kinetic Tree Theory [9]. One example is WAM/BAM [10], which probabilistically evaluates systems modeled with Boolean algebra. Another example is the PREP/KITT [7] code which computes the failure probabilities associated with the fault tree of a system. The minimal cut sets of the system (the smallest sets of components that are sufficient to cause system failure) which are required by KITT,

are provided by the companion PREP. Minimal cut sets are also found by a method and algorithm presented by Garribba et al. [11].

Improvement on all the above methods has been done by Chamow [9]. His method involves the use of directed graph and related matrix methods so as to speedily transform the graphics into corresponding matrix. The major benefit is because the mathematical solutions are readily performed by standard matrix techniques, which can be implemented either manually or by computer. Other computer aided methods were developed by Lapp and Powers [12], Schneewis [13] and Semanderes [14], the last one developing the ELRAFT code for efficient logic reduction analysis of fault trees.

Fuel in a nuclear power plant, is located in the reactor core, the spent fuel pool, the refuelling process area, and the spent fuel shipping cask. By far the largest amount of radioactivity is located in the fuel in the reactor core. The much larger amount of radioactivity that resides in the core, as opposed to other locations is one of the main reason why the bulk of attention in safety of nuclear power plants has been directed towards potential accidents involving only the core. According to Levine and Vesely [15], the refuelling operation and the shipping cask are not important contributors to overall risk, since it is exceedingly unlikely that

the fuel will melt in these situations. They conclude that a potentially large release of radioactivity could involve the fuel in the reactor core or in spent-fuel pool.

For any computation of risk at any of the stages mentioned above, the prerequisite is calculation of the fission product inventories. Clarke [16] in 1972 gave the computer code FISP for comprehensive calculation of fission product inventories. The FISP program produces number densities and activities of fission products for a type of reactor fuel after a given irradiation history. The program considers all the radioactive and stable nuclides between ^{72}Zn and ^{161}Dy to make it very comprehensive. The fission product chains upto six isotopes long, and half lives above 1 sec. are included.

The next stepping stone for consequence assessment was again provided by Clarke [17]. The WEERIE program is essentially for assessing the radiological consequences of airborne effluents from nuclear installation. The integrated effluent is dispersed down wind and PASQUILL weather categories are used. The concentration thus obtained are then used to obtain inhalation doses.

More recent work published along this time is the work of Fryer and Kaiser [18] in 1978. TIRION 2 and TIRION 4 are computer programs for use in nuclear safety studies. The later one is the improved version. It includes a systematic

study of plume rise, several refinements of the meteorological model employed, a flexible approach to the relationship of dose and consequence and an examination of milk ingestion path-way.

Refinement of this work has been carried out by Strachan and Godard [19]. Their model calculates the depletion, by dry deposition and washout to the ground, from a plume of fission products in the atmosphere, and for determining the isotopic composition of resulting gamma and beta activities in the ground deposit at any selected point downwind.

The transportation of radioactive materials requires special attention because of the unique hazard associated with such cargoes. The first major work was attempted in this direction by Leimkuhler [20]. In the absence of substantial experience, specially for shipment of highly radioactive materials, the evaluation of risk is quite arbitrary. Leimkuhler attempts to solve the problem by making it a safety vs. economy problem. He restates the problem as one of cost minimization, where total cost includes both the cost of control and the liability for accident losses.

Generally all accident studies consider radioactive doses to an essentially stationary population along the transportation routes or surrounding the site of an assumed transportation accident that has led to dispersal of

radioactivity. Yadigaroglu [21] stresses, however, that the dose to the traffic passing the fuel-cask-carrying vehicle might also be of importance on busy highways. Similarly, for highway accidents resulting in some radioactivity release, he considers the radioactive dose to the occupants of the vehicles that pass in the proximity of point of release.

Smeed and Jacobs [22] have attempted at the analysis of fatalities both in the developed and developing countries based on the number of road fatalities, vehicles and population. Jacobs was able to highlight the fact that the less developed countries (with lower vehicle ownership) will have the higher fatality rates. This fact does have a bearing at calculating the accident probability in road transport.

More recently, Swindell [23] in 1978 has brought out the importance of packaging and shipping cask design in the transportation of radioactive material. The interest on packaging and transportation was initially evoked in the Geneva IV Conference, by the work of Ferguson and Salmon [24] and Richards [25]. The special issue of report on packaging and transportation highlights the interest created. Devine [26] analyses the Department of Transportation [DOT] Regulation whose objectives is the protection of the general public, transportation workers and property. It emphasises that

approaches towards ensuring safety is to establish rigorous packaging standards to eliminate the possibility of release of contents under adverse conditions or to limit the possible release to that quantity that poses no significant hazard.

1.4 PRESENT WORK:

The present work is an effort to estimate risk to public following an accident involving spent fuel. During the operation of the CANDU reactor, fission products are generated. The spent fuel is then left to 'cool' in the Spent Fuel Storage Pool normally for 120 days but in India the cooling period is around 400 days. In Chapter II, the methodology involved in calculating the fission products in spent fuel, and the decay of activity with time is presented. Chapter III contains description of the fault trees used to compute the accident probabilities in both the SFSP, as well as in transportation of the fuel bundles from the reactor site to the reprocessing plant. Monte-Carlo simulation used to quantify the fault trees and the confidence interval estimate of the TOP event for both spent fuel storage and transportation are given. Chapter IV contains description on the dispersion of the effluent under different weather conditions, and computation of dose due to both gamma and beta rays and final risk. Chapter V contains the results, the conclusions drawn from them and an outline for further work.

CHAPTER II

FISSION PRODUCT INVENTORY

The radiation exposure that the public may receive in the event of an accident is governed by the amount of fission product inventory at that time. The quantity of fission products existing can be accurately estimated from the operating history (power level, time, operating schedules). Fission products include many atomic species in varying amounts and with different physical chemical and radioactive properties.

Though neutron 'fluence' or 'flux-time' are expressed as a measure of exposure the most popular unit of exposure of 'burn-up' used is megawatt days per tonne (MWD/T) of uranium. Another unit frequently used is the atom fraction burn up. This unit is defined as,

$$\beta = \frac{\text{Number of fission product atoms formed}}{\text{Number of fuel atoms in reactor}}$$

These two units are related by the expression for uranium 235 fuel,

$$\text{burn up in MWD/T} = 9.5 \times 10^5 \times \beta.$$

2.1 OPERATING SCHEMES:

The simplest operating scheme is the batch type in which the total fuel loaded in the reactor is discharged at a time. The fuel at the centre of the reactor is exposed to a maximum neutron flux due to the cosine flux shape. In this scheme, the burnup of the fuel is not uniform, and fuel utilisation is poor.

Other schemes to improve the fuel utilisation are (1) Graded irradiation, (2) Steady axial movement of fuel, (3) Salt pepper method. The steady axial movement method is of particular interest since this scheme is utilised in the CANDU reactors. Steady axial movement of fuel is aimed at achieving uniform burn-ups of the fuel bundles in each channel. Fuel bundles in the neighbouring coolant channels are moved in the opposite directions. This scheme is, therefore, called the bidirectional fuelling. The rate of fuel bundle movement is chosen so as to give uniform burn up of all the discharged fuel bundles irrespective of their location in the reactor. A CANDU reactor discharges 8 fuel bundles per day.

The burnup achieved in a scheme having bidirectional steady axial movement of fuel is far higher than the batch type irradiation scheme. Burnup for a typical Rajasthan reactor is 7000 MWD/T though theoretically a burnup of 10,000 MWD/T is possible, due to plutonium 239 build up, and its recycle.

2.2 FISSION PRODUCT COMPOSITION OF A TYPICAL CANDU REACTOR:

For the analysis of the spent fuel composition, differential equations governing the various fission product pairs in the fuel element are given below.

$$\frac{d N_F (25)}{dt} = N_{25} \sigma_{25} \phi / (1 + \alpha_{25}) \quad (2.1)$$

The solution to the above differential gives the rate of growth of the fission product pairs from uranium 235, as follows:

$$N_F (25) = N_{25} (0) (1 - \exp (-\sigma_{25} \theta) / (1 + \alpha_{25})) \quad (2.2)$$

The plutonium 239 fission contribute the following concentration,

$$N_F (49) = \sigma_{49} / (1 + \alpha_{49}) [(C_1 \theta + C_2 (1 - \exp (-\sigma_{25} \theta)) / \sigma_{25} - (C_1 + C_2) (1 - \exp (-\sigma_{49} \gamma \theta)) / \sigma_{49} \gamma] \quad (2.3)$$

Plutonium 240 is not fissionable. Plutonium 241 yields a concentration of fission products pairs governed by,

$$N_F(41) = [\sigma_{41} (1 + \alpha_{41})] [C_6 \theta + C_7 (1 - \exp (-\sigma_{25} \theta)) / \sigma_{25} + (C_8 / \sigma_{49} \gamma) (1 - \exp (-\sigma_{49} \gamma \theta)) + (C_9 / \sigma_{40}) (1 - \exp (-\sigma_{40} \theta) - (C_6 + C_7 + C_8 + C_9) (1 - \exp (-\sigma_{41} \theta)) / \sigma_{41}] \quad (2.4)$$

Fast fission in uranium 238 will produce fission products. The concentration of fission products formed in fast fission of uranium -238 is governed by,

$$N_F(28) = (\epsilon-1)/(\gamma_{28}-1) [\gamma_{25} N_F(25) + \gamma_{49} N_F(49) + \gamma_{41} N_F(41)] \quad (2.5)$$

Burn-up of fuel is related to fission product pairs by,

$$\beta = [N_F(25) + N_F(49) + N_F(41) + N_F(28)] / [N_{25}(0) + N_{28}(0)] \quad (2.6)$$

where the constants occurring in equation (2.3) and (2.4) are,

$$\gamma = 1 - \eta_{49} \epsilon P_1 (1-p) \quad (2.7)$$

$$C_1 = N_{28}(0) \sigma_{28}/\sigma_{49} \gamma \quad (2.8)$$

$$C_2 = \eta_{25} \epsilon P_1 (1-p) N_{25}(0) \sigma_{25}/(\sigma_{49}\gamma - \sigma_{25}) \quad (2.9)$$

$$C_3 = N_{28}(0) \sigma_{28} \alpha_{49}/\sigma_{40} \gamma (1 + \alpha_{49}) \quad (2.10)$$

$$C_4 = C_2 \sigma_{49} \alpha_{49}/(\sigma_{40} - \sigma_{25}) (1 + \alpha_{49}) \quad (2.11)$$

$$C_5 = C_3 \sigma_{40}/(\sigma_{49}\gamma - \sigma_{40}) + C_4(\sigma_{40} - \sigma_{25})/(\sigma_{49}\gamma - \sigma_{40}) \quad (2.12)$$

$$C_6 = C_3 \sigma_{40}/\sigma_{41} \quad (2.13)$$

$$C_7 = C_4 \sigma_{40}/(\sigma_{41} - \sigma_{25}) \quad (2.14)$$

$$C_8 = C_5 \sigma_{40}/(\sigma_{41} - \sigma_{49} \gamma) \quad (2.15)$$

$$C_9 = (C_3 + C_4 + C_5) \sigma_{40}/(\sigma_{40} - \sigma_{41}) \quad (2.16)$$

The fission products considered in the study along with their half lives is listed in Table 2.2. The yields from uranium - 235, Plutonium - 239, and the daughter fission products are listed in Appendix A.

2.3 COMPUTATION DETAILS:

For the computation of the fission product, the following data of a typical CANDU reactor (RAPP) are used.

1. Thermal utilization factor, $f = 0.964$
2. Resonance escape probability $p = 0.9309$
3. Fast fission factor $\epsilon = 1.027$
4. Prompt neutrons for every neutron absorbed
 - $\eta_{25} = 2.064$
 - $\eta_{49} = 1.9845$
 - $\eta_{41} = 2.223$
5. Neutron produced per fission reaction
 - $\nu_{25} = 2.426$
 - $\nu_{49} = 2.893$
 - $\nu_{41} = 2.317$
6. Mean neutron temperature $T = 200^{\circ}\text{C}$
7. Average burnup $\text{MWD/T} = 7000$
8. The average cross section, to be used, are corrected by use of Westcott's method. Westcott substituted the detailed values of cross section and numerically performed the integration to obtain the average for a non $1/v$ type of cross section, resulting in.

$$\bar{\sigma} = \sigma_0 (g + rs) \quad (2.17)$$

where g and s represent the non $1/v$ and spectrum parameters. The g and s parameters are tabulated in Table 2.1. The parameter r is a measure of the proportion of epithermal neutrons in the reactor spectrum. For CANDU type reactors $r = 0.03$ in moderator and $r = 0.07$ in the fuel. The average cross-section is obtained by using the following formula,

$$\bar{\sigma} = \tilde{\sigma} (\sqrt{\pi}/2) (T_0/T)^{\frac{1}{2}} \quad (2.18)$$

Using values from Table 2.1,

$$g + rs = 0.9482 \quad \text{for all isotopes of uranium}$$

$$g + rs = 1.6760 \quad \text{for plutonium}$$

Equation 2.23 becomes,

$$\bar{\sigma} = 0.6518795 \tilde{\sigma}$$

The average cross sections calculated is listed below:

Isotope	$\sigma_{20^\circ\text{C}}$ (barns)	σ_a , non $1/v$ type	$\sigma_{a,200^\circ\text{C}}$ (barns)	$\alpha = \sigma_c/\sigma_f$
U^{235}	$\sigma_f = 582$ $\sigma_c = 112$	694	429	0.1924
U^{236}	7	7	4.3	
U^{238}	2.71	2.71	1.66	
P_u^{239}	$\sigma_f = 746$ $\sigma_c = 280$	1026	1121	0.3753
P_u^{240}	295	295	322	
P_u^{241}	$\sigma_f = 1025$ $\sigma_c = 375$	1400	1530	0.3659

A total of 79 fission products were considered, as listed in Table 2.2, and the activities at discharge from reactor as well as the decay of the activity at various time intervals was computed. The unit of activity used is curie where one curie is defined as 3.7×10^{10} disintegrations/sec. The activity of a radioactive sample is equal to the number of disintegration per second.

$$\text{Activity } A = N \lambda \text{ disintegration/sec.}$$

$$= \frac{0.693 N}{T_{\frac{1}{2}}} \text{ disintegration/sec.}$$

where $T_{\frac{1}{2}}$ is the half life of the particular fission product. $T_{\frac{1}{2}}$ for all the 79 fission product is listed in Table 2.2.

The computer program prepared to compute the activities of the fission products at discharge, also computes the activities of the products as they 'cool down'. The production of daughters of the fission products is also taken into consideration. The general solution for the build-up of an amount N_i of the i -th member of the fission product chain,

$$N_i = \lambda_1 \lambda_2 \dots \lambda_{i-1} N_f \sigma_f \phi y_1 \times \sum_{j=1}^i \frac{1 - e^{-\mu_j t}}{\mu_j \prod_{k \neq j} (\mu_k - \mu_j)} \quad (2.19)$$

where μ is the total removal -rate constant and is given by,

$$\mu_i = \lambda_i + f_i + \phi \sigma_i$$

where,

N_f = number of atoms of fissionable material
in reactor

σ_f = fission cross section, cm^2

ϕ = average neutron flux, $\text{cm}^2 \text{sec}^{-1}$

y_1 = yield, atoms of fission product formed directly
from fission per atom of fuel fissioned.

λ = radioactive decay rate constant.

A wide range for cooling time has been taken to analyse the activity changes. The time, as also given in Appendix A, are 1 hour, 5 hour, 10 hour, 25 hour, 100 hour, 10 days, 25 days, 50 days, 80 days, 120 days, 250 days, 1 year and 2 year. The activities are plotted as a function of time in Figure 2.1. The activity decay curve can be divided into four ranges. The classification of the fission products is primarily done to ease the calculations and later analysis. It becomes easy to handle four groups of fission products than to handle 79 fission products individually. The classes can be named as very rapidly saturating (average half life = 7.9481×10^{-4} yrs.), rapidly saturating (average half life = 0.0161 year), slowly saturating

(average half life = 0.2402 year) and very slowly saturating (average half life = 19.77 year) fission products. The first group contains the first 39 fission products, the second group contains the next 23 fission products, the third group containing the next 9 fission product, and the last group having the last 8 fission products.

Table 2.1: Westcott g and s parameters.

Element	T°C	Absorption			Fission		
		g	s	($\gamma=.07$) $\bar{\sigma}$	g	s	($\gamma=.07$) $\bar{\sigma}$
U 233	20	.9983	1.292	629.3	1.0003	1.222	570.1
$\sigma_{ao} = 578$	60	.9976	1.380	632.4	1.0007	1.304	571.7
$\sigma_{fo} = 525$	100	.9972	1.462	635.5	1.0011	1.380	576.3
	160	.9971	1.576	640.1	1.0019	1.474	580.7
	220	.9975	1.687	644.6	1.0029	1.587	584.8
	280	.9984	1.781	649.1	1.0040	1.682	588.9
	360	1.0000	1.906	655.1	1.0058	1.801	594.2
	450	1.0024	2.041	662.0	1.0081	1.929	600.2
	540	1.0052	2.172	668.9	1.0108	2.053	605.2
	600	1.0072	2.259	673.6	1.0128	2.135	610.2
U 235	20	.9780	.0497	670.4	.9759	-.0502	561.1
$\sigma_{ao} = 683.04$	60	.9689	.0725	665.3	.9665	-.0324	566.4
$\sigma_{fo} = 577.01$	100	.9610	.0923	660.8	.9581	-.0169	552.2
	160	.9511	.1159	655.2	.9473	.0017	546.7
	220	.9433	.1323	650.7	.9384	.0144	542.0
	280	.9373	.1425	647.1	.9312	.0217	538.2
	360	.9316	.1492	643.4	.9238	.0251	534.0
	450	.9273	.1528	640.7	.9177	.0249	530.5
	540	.9244	.1607	639.0	.9133	.0254	528.0
	600	.9229	.1639	638.2	.9108	1.0274	526.7
Pu 239	20	1.0723	2.960	1316.8	1.0487	2.329	899.9
$\sigma_{ao} = 1029.1$	60	1.1117	3.085	1366.2	1.0777	2.432	926.2
$\sigma_{fo} = 742.15$	100	1.1611	3.130	1420.4	1.1150	2.471	955.9
	160	1.2582	3.046	1514.4	1.1898	2.407	1008.1
	220	1.3836	2.778	1624.1	1.2880	2.192	1069.8
	280	1.5345	2.335	1747.4	1.4071	1.836	1139.6
	360	1.7658	1.540	1928.2	1.5910	1.195	1242.8
	450	2.0505	0.536	2148.8	1.8182	0.384	1369.3
	540	2.3419	-.400	2381.3	2.0514	-.376	1503.1
	600	2.5307	-.070	2570.0	2.2037	-.008	1503.5

Table 2.2: Fission Product Inventory.

Fission Product Name	Half Life (Yr.)	Activity (Curies)	Fission Product Name	Half Life (Yr.)	Activity (Curie)
Se ⁸⁷	0.507×10^{-6}	0.139×10^8	Ga ⁷³	0.559×10^{-3}	0.694
Se ⁸⁵	0.123×10^{-6}	0.314×10^7	I ¹³⁵	0.764×10^{-3}	0.510×10^5
I ¹³⁶	0.272×10^{-5}	0.637×10^7	Xe ¹³⁵	0.105×10^{-2}	0.211×10^5
Sn ¹³⁰	0.494×10^{-5}	0.142×10^7	Sr ⁹¹	0.111×10^{-2}	0.250×10^5
Br ^{84m}	0.114×10^{-4}	0.587×10^4	Y ⁹³	0.117×10^{-2}	0.183×10^5
Ga ⁷⁴	0.148×10^{-4}	0.832×10^2	Ge ⁷⁷	0.128×10^{-2}	8.48
As ⁷⁹	0.171×10^{-4}	0.115×10^5	I ¹³⁰	0.143×10^{-2}	1.22
Se ^{81m}	0.35×10^{-4}	0.141×10^5	Pd ¹⁰⁹	0.152×10^{-2}	0.287×10^4
Rh ¹⁰⁷	0.418×10^{-4}	0.160×10^5	Eu ¹⁵⁷	0.175×10^{-2}	15.666
Sm ¹⁵⁵	0.456×10^{-4}	0.255×10^4	Zr ⁹⁷	0.194×10^{-2}	0.194×10^5
Se ⁸³	0.475×10^{-4}	0.163×10^5	Gd ¹⁵⁹	0.205×10^{-2}	1.8389
I ¹²⁸	0.475×10^{-4}	0.222×10^1	I ¹³³	0.237×10^{-2}	0.981×10^4
Br ⁸⁴	0.605×10^{-4}	0.536×10^5	Pd ¹¹²	0.239×10^{-2}	0.168×10^3
Nb ⁹⁸	0.989×10^{-4}	0.848×10^5	Nb ⁹⁶	0.262×10^{-2}	5.0255
I ¹³⁴	0.998×10^{-4}	0.275×10^6	Sn ¹²¹	0.313×10^{-2}	58.881
Sn ¹²⁸	0.103×10^{-3}	0.120×10^5	Ce ¹⁴³	0.376×10^{-2}	0.965×10^4
Se ⁸¹	0.108×10^{-3}	0.273×10^3	Br ⁸²	0.409×10^{-2}	0.0344
Eu ¹⁵⁸	0.114×10^{-3}	0.618×10^2	Rh ¹⁰⁵	0.410×10^{-2}	0.291×10^4
Ba ¹³⁹	0.157×10^{-3}	0.260×10^6	As ⁷⁷	0.441×10^{-2}	6.6339
As ⁷⁸	0.173×10^{-3}	0.407×10^3	Sm ¹⁷³	0.536×10^{-2}	0.310×10^3
Ge ⁷⁸	0.239×10^{-3}	0.294×10^3	Zn ⁷²	0.536×10^{-2}	0.0791
Br ⁸³	0.273×10^{-3}	0.751×10^4	Cd ¹¹⁵	0.605×10^{-2}	7.587
Sr ⁹²	0.308×10^{-3}	0.607×10^5	Mo ⁹⁹	0.759×10^{-2}	0.528×10^4
Cd ¹¹⁷	0.342×10^{-3}	0.113×10^3	Te ¹³²	0.878×10^{-2}	0.366×10^4
La ¹⁴¹	0.433×10^{-3}	0.923×10^5	Dy ¹⁶⁶	0.936×10^{-2}	0.0222
Ru ¹⁰⁵	0.507×10^{-3}	0.625×10^4	Sb ¹²⁷	0.103×10^{-1}	0.159×10^3

contd...

Fission Product Name	Half Life (Yr.)	Activity (Curie)	Fission Product Name	Half Life (Yr.)	Activity (Curie)
Xe ¹³³	0.01443	0.304x10 ⁴	Cd ¹¹⁵	0.1178	0.1601x10 ³
Tb ¹⁶¹	0.0189	0.6469	Sr ⁸⁹	0.1383	0.222x10 ³
Ag ¹¹¹	0.0208	37.099	Zr ⁹⁵	0.1780	0.4296
I ¹³¹	0.0228	0.102x10 ⁴	Te ¹²⁷	0.2876	0.0123
Sn ¹²⁵	0.0263	10.024	Sn ¹²³	0.3726	42.769
Nd ¹⁴⁷	0.03041	535.36	Ce ¹⁴⁴	0.7671	15.205
Ba ¹⁴⁰	0.0350	0.111x10 ⁴	Ru ¹⁰⁶	1.01	0.037
Cs ¹³⁶	0.0356	10.146	Sb ¹²⁵	2.001	2.288
Eu ¹⁵⁶	0.0421	9.167	Pm ¹⁴⁷	2.600	0.0291
Rb ⁸⁶	0.0509	0.339x10 ⁻²	Eu ¹⁵⁵	4.00	0.1343
Ce ¹⁴¹	0.0904	0.407x10 ³	Kr ⁸⁵	10.6	0.913x10 ²
Te ¹³⁹	0.1013	12.19x10 ²	Sr ⁹⁰	28.0	0.974
Ru ¹⁰³	0.1087	0.257x10 ³	Cs ¹³⁷	30.0	1.401
			Sm ¹⁵¹	80.0	0.0508

Time after discharge	Activity (Curie)	Time after discharge	Activity (Curie)
1 hour	0.339x10 ¹⁰	50 days	0.5132x10 ⁶
5 hour	0.5503x10 ⁹	80 days	0.2921x10 ⁶
10 hour	0.1999x10 ⁹	120 days	0.145 x 10 ⁶
25 hour	0.3869x10 ⁸	280 days	0.1269x10 ⁵
50 hour	0.9848x10 ⁷	1 year	0.4509x10 ⁴
100 hour	0.3185x10 ⁷	2 year	0.6541x10 ³
10 days	0.1425x10 ⁷	5 year	0.3469x10 ³
25 days	0.8689x10 ⁶		

CHAPTER III

FAULT TREE CONSTRUCTION AND QUANTIFICATION

A systematic methodology for the construction of fault tree is presented in this chapter. In order to construct fault trees for any system, information is required both to describe the system itself, and the operation of the specific components within the system. It is desired to develop component models which are independent of any specific system, and which depend only upon a set of inputs from other components within the system. With a group of such component models available, the description of the system configuration simply requires a specification of the interconnections of inputs and outputs of various components, along with a TOP event in order to begin construction of the tree.

Figure 3.1 gives the logic symbolism and description of events used in the construction of fault trees. Boolean algebra used for interconnection of the various inputs to the top event or describing functional relationship is given in Appendix B.

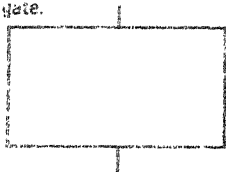
3.1 FAULT TREE FOR SPENT FUEL STORAGE POOL:

The fault tree construction process begins with the definition of the undesirable event. Once the undesirable

FAULT TREE SYMBOLISM

EVENT REPRESENTATIONS

The rectangle identifies an event that results from the combination of fault events through the input logic gate.



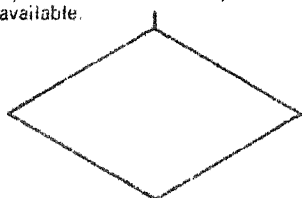
The circle describes a basic fault event that requires no further development. Frequency and mode of failure of items so identified are derived from empirical data.



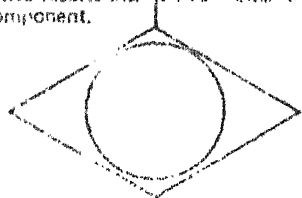
The triangles are used as transfer symbols. A line from the apex of the triangle indicates a transfer in and a line from the side or bottom denotes a transfer out.



The diamond describes a fault event that is considered basic in a given fault tree. The possible causes of the event are not developed further because the event is of insufficient consequence or the necessary information is unavailable.



The circle within a diamond indicates a subtree exists, but that subtree was evaluated separately and the quantitative results inserted as though a component.



The house is used as a switch to include or eliminate parts of the fault tree as those parts may or may not apply to certain situations.

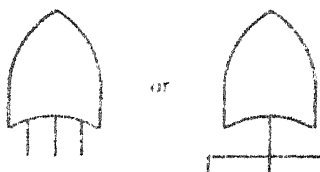


LOGIC REPRESENTATIONS

AND gate describes the logical operation whereby the coexistence of all input events is required to produce the output event.



OR gate defines the situation whereby the output event will exist if one or more of the input events exists.



INHIBIT gates describe a causal relationship between one fault and another. The input event directly produces the output event if the indicated condition is satisfied. The conditional input defines a state of the system that permits the fault sequence to occur, and may be either normal to the system or result from failures.

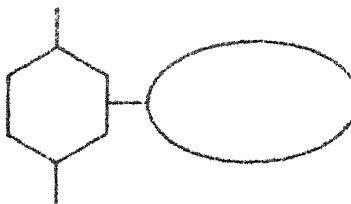


Figure 3.1: Standard Fault Tree Logic and Event Symbolism

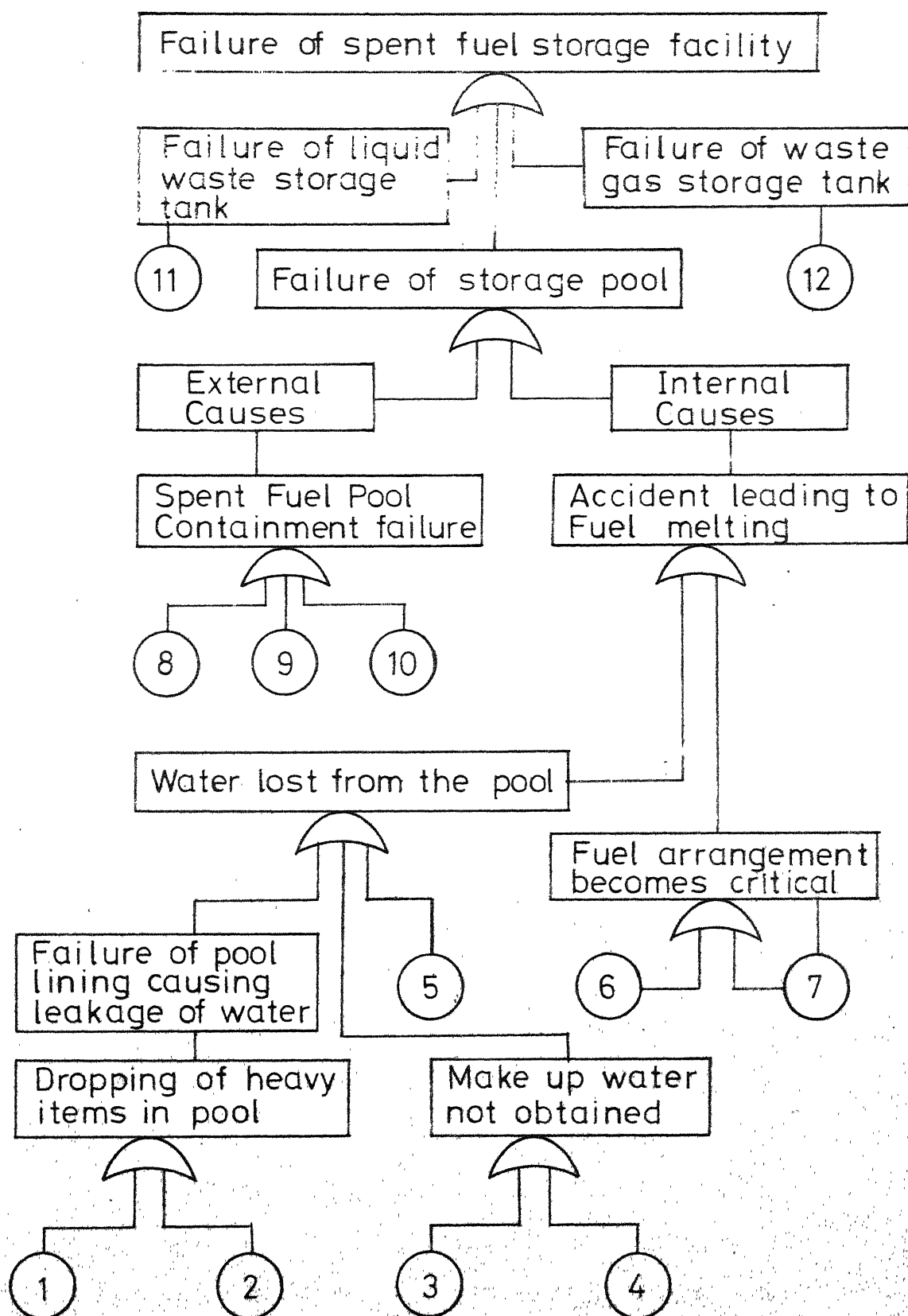


Fig. 3-2 Fault tree of SFSP

event is selected for the system, it becomes the TOP event on the fault tree. For our case the TOP event is the 'failure of spent fuel storage facility.'

Figure 3.2 shows the tree of potential accidents considered for the spentful pool. The accident mode is clearly divided into two, first the accidents that can be initiated by internal plant failures and second, the external forces that can potentially cause accidents.

Release of radioactivity from stored spent-fuel due to internal failures can potentially result from heat imbalances causing melting of stored fuel or from mechanical damage to the fuel assemblies causing release of gap activity. Heat imbalance can result from:

- 1) loss of coolant water from the spent-fuel storage-pool,
- 2) loss of the capacity to remove heat from the pool water, which would lead to boiling away of the pool water.
- 3) an increase in the heat generation rate in the pool because the configuration had been altered into a critical array, again leading to the boil-off of pool water.

The external causes resulting in failure of spent fuel storage facility will be through failure of SFSP containment failure. The external causes considered here are due to 1) damage due to seismic activity, 2) damage by tornado

and 3) damage due to turbine missiles.

The 12 basic units of the fault tree of SFSP as shown in Figure 3.2 are tabulated in Table 3.1 with their failure probabilities per Reactor-Year. The source of data so tabulated in Table 3.1 is taken from WASH-1400, Appendix 3, 4 and 5 [27,28,29].

Table 3.1: Fault Tree for Spent Fuel Storage Pool

Basic unit of fault tree	Accident mode	Probability of occurrence per reactor-year
1	Dropping of casks due to crane failure	3×10^{-6}
2	Human error	10^{-3}
3	Make-up water unavailable	10^{-3}
4	Make-up Water need not recognised	10^{-5}
5	Water lost from pool due to inadvertent opening of valves	10^{-3}
6	Dropping of heavy items (failure of crane)	3×10^{-6}
7	Seismic forces resulting in fuel arrangement to become critical	3×10^{-8}
8	Turbine missiles	10^{-6}
9	Tornado	10^{-9}
10	Damage due to seismic forces	3×10^{-8}
11	Liquid waste storage tank rupture	10^{-2}
12	Waste gas storage tank rupture	10^{-2}

3.2 COMMON CAUSE FAILURE IN SFSP TREE:

The successful definition of common cause failures is necessary to help ensure that all the important contributing accident sequences have been defined and that the probabilities of occurrence of the sequences adequately predicted. The identification of common-cause failure therefore is an integral part of construction of fault tree.

Once the TOP event has been defined and the construction of fault trees started common cause failures and dependencies are incorporated into the fault trees and their quantification in the following five ways, as was also done in WASH-1400, Appendix 2 [7].

1. The fault trees were constructed to meet the context prescribed by TOP event, the fault trees are now conditional.
2. The fault-trees identify components that are common to multiple systems appearing in an accident sequence,
3. Each fault tree is developed to a detailed component level in order to locate single component failures and potential common-cause failures deep within a system.
4. Human failures are explicitly included in fault trees, and dependencies between human failures were also included in the fault-tree quantification.
5. Evaluations, including bounding and sensitivity studies, were performed to determine the possible impacts from common-cause failures.

In the SFSP fault tree, it is possible to identify common causes and dependencies that are due to not only hardware (crane failure leading to dropping of heavy items in the pool) but external cause (Damage due to Seismic activity) and human failures. Human failure is included in the fault tree wherever the operator interfaced with a component or subsystem and could cause failure. Historically, human failures were often not included in fault trees; this was particularly true when fault tree was constructed at the conceptual design stage of the system, where much information was not available.

In the SFSP fault tree, the common mode failures are:

- (i) Component 1, dropping of casks resulting in damage to pool lining and component 6, dropping of heavy objects to cause fuel arrangements to become critical. The common cause here is failure of the crane.
- (ii) Component 2, dropping of some heavy items resulting in damage to pool lining and component 5, inadvertent opening of valves in SFSP, resulting in loss of water from the pool. The common cause here is human error.
- (iii) Component 7, fuel arrangement becomes critical due to seismic forces and component 10, spent fuel pool containment failure due to seismic activity.

3.3 FAULT TREE OF TYPICAL CANDU SPENT FUEL TRANSPORTATION ACCIDENT MODEL:

A single consignment for shipment in a truck consists of 250 fuel bundles out of the 3000 fuel bundles in core. So a total of 12 shipment have to be done to transport one core load of the fuel bundles.

For each shipment, the undesirable 'TOP' event is 'severe/fatal accident of motor vehicle.' Transport Research Division, Ministry of Shipping and Transport, Government of India in their publication [32] define severe/fatal accident as an accident in which one or more person (occupants of motor vehicle) were killed outright or died within 30 days as a result of accident.

Figure 3.3 shows the tree of potential accidents considered for the transportation accident model. Like the SFSP accident describe in Section 3.1, the tree can be divided into two, first the accidents due to inherent failure of some critical item in the motor vehicle itself and second, the external causes which ultimately result in accident. The internal causes considered are defective brakes, defective steering, insufficient or no light, puncture or burst tyre, overloading and other serious mechanical defects. The external causes resulting in failure considered are defective road surface, collision with trains, collision with tree and other

stationary objects, and wrong human judgement both on the part of driver of motor vehicle and driver of vehicles other than the considered motor vehicle.

The 11 basic units of the fault tree of the transportation model as shown in Figure 3.3 are tabulated in Table 3.2 with their failure probabilities. The source of data tabulated in Table 3.2 are the Ministry of Shipping and Transport, Govt. of India publication of Transport Statistics [32,33,34]. All the data belong to the basic year of 1974-75.

Table 3.2: Fault Tree of Transportation Accident Model.

Basic unit of fault tree	Accident mode	Probability of occurrence per kilometer travel
1	Wrong human judgement, fault of driver of motor vehicle	6.45×10^{-8}
2	Wrong human judgement, fault of driver of vehicles, other than the motor vehicle considered	8.1×10^{-10}
3	Collision with tree/stationary objects	2.33×10^{-8}
4	Collision with train	2.15×10^{-10}
5	Defective road surface	3.3×10^{-10}
6	Insufficient or no light	1.2×10^{-10}
7	Puncture or burst tyre	2.3×10^{-10}
8	Serious mechanical defects other than defective steering or brake	4.86×10^{-10}
9	Defective steering	2.5×10^{-10}
10	Defective brakes	5.45×10^{-10}
11	Overloading	8.9×10^{-10}

3.4 GENERAL APPROACH TO FAULT TREE QUANTIFICATION:

The fault trees in themselves are qualitative in nature, but they provide a frame work for quantitative probabilistic analysis. Binary modeling of components in a failed state is a characteristic feature of fault tree analysis. To facilitate the quantitative analysis it is convenient to represent the fault tree in mathematical form, and Boolean algebra is an appropriate tool for this purpose.

Given a fault tree the top event T can be expressed as Boolean functions of primary events E_i such as

$$T = f(E_1, E_2, E_3, \dots) \quad (3.1)$$

For fault trees this can be written as,

$$T = M_1 + M_2 + M_3 + \dots \quad (3.2)$$

The events M_i are secondary events consisting of intersections of primary events,

$$M_i = \sum_{k=1}^m C_{i_k}$$

where no M_i is a subset of another M_j . With the expression in this form, the M_i are termed the critical paths or minimal cut sets of the fault tree.

The probability of T for small probability events can be written as,

$$P(T) = \sum_{i=1}^n P(M_i) \quad (3.4)$$

and if primary events are independent then,

$$P(M_i) = \sum_{k=1}^m P(C_{i_k}) \quad (3.5)$$

In case, common mode failures exist, they are included and quantified, as explained in Appendix B.

3.5 DATA TREATMENT:

The quantification of fault trees involve two basic approaches: point estimation or the random variable evaluation. To produce a meaningful point estimate, the input data of components should be highly accurate. In actual practice, however, extensive failure rate data to back point estimation are not usually available.

The alternative approach is to use the random variable technique. Although in use before, this method became very popular with the publication of WASH-1400 [3], where it found extensive use.

To express uncertainty in failure rate data a lognormal distribution was fitted and the median as well as 5 and 95 percentile point values are estimated. The lognormal distribution was chosen because of large variability in data, its flexibility, its consistency with reliability data and because of its property that for a complex system, the

probability for the top event is the simple additive of the component probabilities. These justifications are elaborated in Appendix B.

When fault tree is simple, statistical distribution algebra can be used to compute the location parameter and the variance of TOP event unavailability. When fault tree is complicated it is convenient to use 'MONTE-CARLO' simulation of fault tree.

3.6 POINT ESTIMATION FOR TOP EVENT:

Using equation 3.5, the top event for the SFSP fault tree is denoted by,

$$\begin{aligned} P(T) = & X(1) + X(2) + X(3) + X(4) + X(5) + X(6) + X(7) \\ & + X(8) + X(9) + X(10) + X(11) + X(12) + X(13) \\ & + X(14) + X(15) \end{aligned} \quad (3.6)$$

The first 12 probabilities are those listed in Table 3.1, and the probabilities $X(13)$, $X(14)$ and $X(15)$ are contribution due to common mode failures. Generally these are second order effects. The computation method for common mode failures, which has been dealt in Appendix B, can be written as,

$$X(13) = [X(1).X(6).X(1)]^{\frac{1}{2}} = 6 \times 10^{-14} \quad (3.7)$$

$$X(14) = [X(2).X(5).X(2)]^{\frac{1}{2}} = 3 \times 10^{-5} \quad (3.8)$$

$$X(15) = [X(7).X(10).X(7)]^{\frac{1}{2}} = 5 \times 10^{-12} \quad (3.9)$$

The point estimation of SFSP comes to be,

$$P(T) = 0.02405 \quad (3.10)$$

The top event for the transportation model is denoted by the Boolean expression,

$$P(T) = X(1) + X(2) + X(3) + X(4) + X(5) + X(6) + X(7) \\ + X(8) + X(9) + X(10) + X(11) \quad (3.12)$$

The point estimation of transportation model comes to be,

$$P(T) = 0.9249 \times 10^{-7}$$

Point estimates for accident probabilities of transportation can be obtained by direct evaluation of road fatalities which can be established by the use of Smeeds formula [22],

$$F/V = 0.0003 (V/P)^{-2/3}$$

and Jacobs formula [22]

$$F/V = 0.00077 (V/P)^{-2/3}$$

where,

F = road fatalities,

V = number of vehicles,

P = population.

Jacobs formula is of more interest as it is valid for developing countries like India whereas Smeeds formula represents developed countries.

3.7 MONTE CARLO SIMULATION:

For each fault appearing in the Boolean expression for a system fault tree, a particular failure rate is obtained by a random sampling of the appropriate log normal distribution. These failure rates are then used to compute a value for the top-event failure. This process is repeated for a large number of trials to obtain a distribution of top event characteristics. Computer code SAMPLE [7] is used for Monte-Carlo simulation.

The SAMPLE program uses Monte-Carlo sampling to obtain the mean, standard deviation, probability range and distribution for a function,

$$Y = f(X_1, X_2, \dots, X_n) \quad (3.15)$$

Given a function $Y = f(X_1, X_2, \dots, X_n)$, values of the location and dispersion parameters of the independent variables, and the input distribution, SAMPLE obtains a Monte Carlo sampling, X_1, X_2, \dots, X_n from the input and evaluates the function Y . The sampling is repeated N times and the resultant estimates of Y are ordered in ascending value $y_1 \quad y_2 \quad \dots \quad y_n$ to obtain the limits (percentiles) of the distribution of Y .

The input parameters to the SAMPLE program are N (no. of trial runs), median values associated with the 90 percent

probability ranges. The upper and lower bounds of the 90 percent interval are determined by the 95 percent and 5 percent percentiles of the log normal probability distribution. The top-event distribution is approximated by a log-normal by assigning 90 percent error factor which corresponds to the 90 percent interval. The error factor is defined as the ratio of the end point to the median value. The program prints the result in terms of empirical probability percentiles. The 50 percent value is the median of the distribution, and the 5 percent and 95 percent values are bounds of 90 percent probability interval. The program also generates a frequency function on which the median value and the range can be indicated.

CHAPTER IV

ATMOSPHERIC DISPERSION AND DOSIMETRIC CALCULATION

4.1 ATMOSPHERIC DISPERSION:

As the plume from a site of accident involving release of radioactivity is carried away from the site of accident by wind, atmospheric diffusion would be continually acting to disperse the contaminants at a rate depending on the wind speed, thermal stability, and the characteristics of the terrain.

The most important meteorological parameter for this analysis is the atmospheric transport vector, or wind. For the instantaneous source case, it will determine how quickly the effluent puff is carried past; for the continuous source case, the wind will determine the direction of effluent travel, and the effluent dilution. Radioactive effluents behave essentially the same way as the smoke plume.

Let 'CHI' be the concentration of some effluent as a function of space and time. As is proved in Appendix C, the effluent concentration 'CHI' in curies/m² is Gaussian in nature, with standard deviations σ_y and σ_z . Assuming that the effluent is emitted at an altitude h from ground level, into an infinite atmosphere, the largest value of 'CHI' or

the concentration of effluent at ground level is given by,

$$'CHI' = \frac{Q}{\pi \bar{v} \sigma_y \sigma_z} \exp \left(- \frac{h^2}{2\sigma_z^2} \right) \quad (4.1)$$

where σ_y , σ_z and h are as described before and \bar{v} is the wind transport velocity.

Equation (4.1) does not take into account that the effluent may be radioactive and may decay while being dispersed. This is handled by replacing Q by $Q \exp(-\lambda t)$ where λ is the decay constant, and t is the time from emission to observation. Therefore, equation (4.1) can be written as,

$$'CHI' = \frac{Q}{\pi \bar{v} \sigma_y \sigma_z} \exp \left[- \left(\lambda t + \frac{h^2}{2\sigma_z^2} \right) \right]$$

or

$$'CHI' = \frac{Q}{\pi \bar{v} \sigma_y \sigma_z} \exp \left[- \left(\frac{x}{\bar{v}} + \frac{h^2}{2\sigma_z^2} \right) \right] \quad (4.2)$$

The standard deviations σ_y and σ_z are called the horizontal and vertical dispersion coefficients and depend on atmospheric condition.

4.2 PASQUILL CONDITIONS:

To use equations 4.1 and 4.2, σ_y and σ_z must be known as a function of distance from the source. Working from experimental data, Pasquill obtained a set of curves for σ_y and σ_z for six different atmospheric stability conditions. They are given in Figures 4.1 and 4.2. A seventh

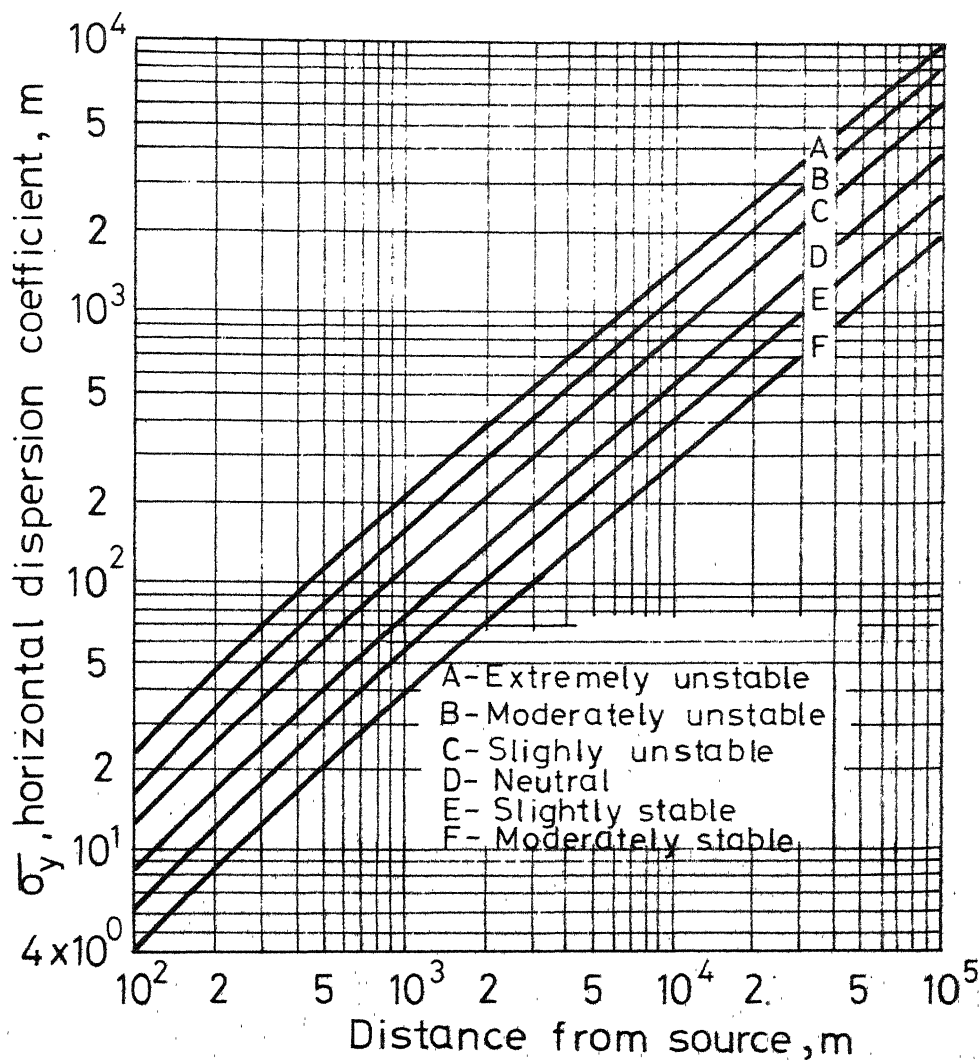


Fig.4.1 Horizontal dispersion coefficient σ_y as a function of distance from source for the various Pasquill conditions

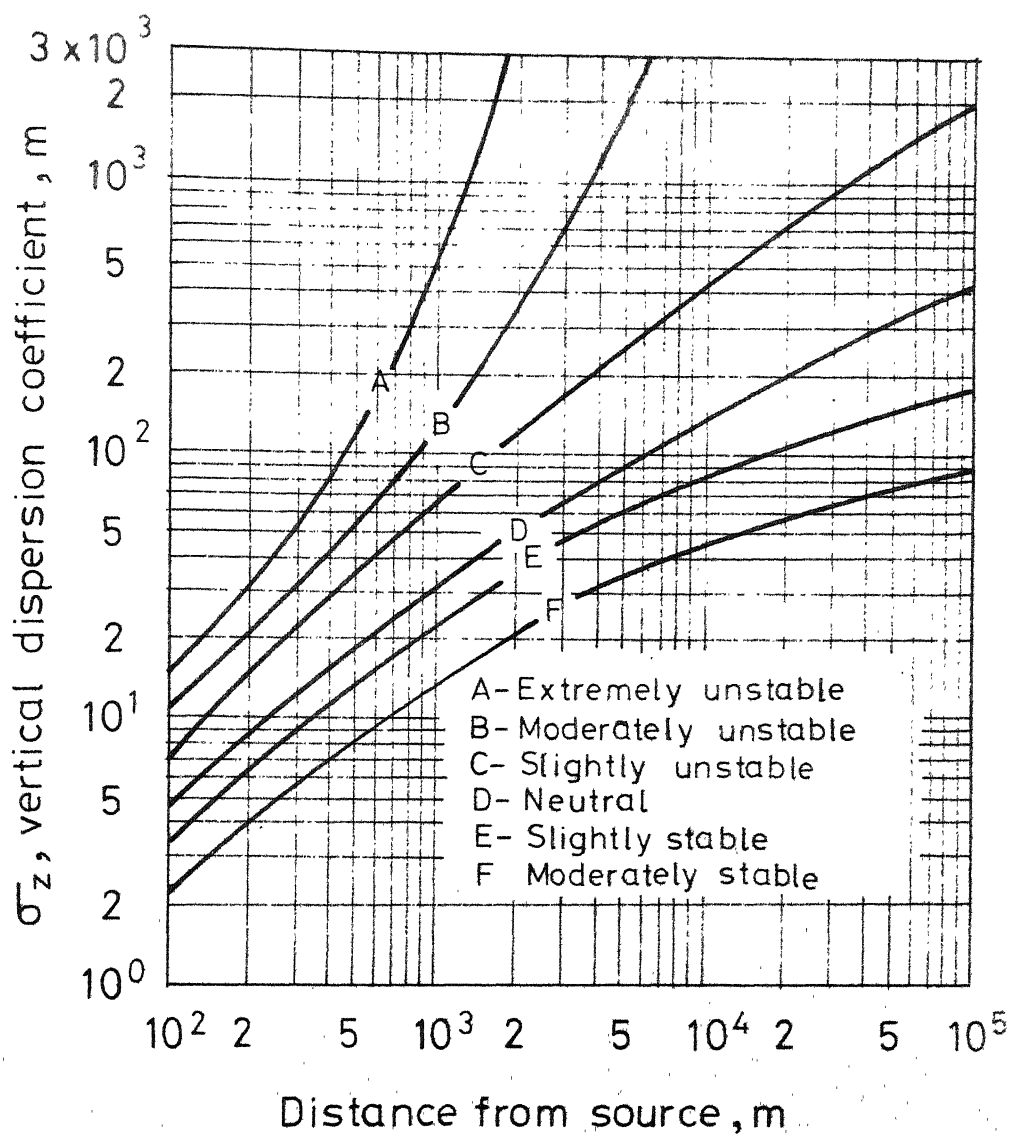


Fig. 4.2 Vertical dispersion coefficient σ_z as a function of distance from source for the various Pasquill conditions.

stability condition, type G, can be approximated by

$$\sigma_z(G) = \frac{3}{5} \sigma_z(F)$$

$$\sigma_y(G) = \frac{2}{3} \sigma_y(F)$$

The seven stability categories A-G are qualitatively called as the extremely unstable, moderately unstable, slightly unstable, neutral, slightly stable, moderately stable, and extremely stable categories. The variation of the atmospheric stability categories during day and night and with wind velocity is given below

Wind speed (m/sec)	Day			Night	
	<u>Incoming Solar Radiation</u>			<u>Cloud Cover</u>	
	Strong	Moderate	Slight	Moderately overcast	Mostly clear
Less than 2	A	A-B	B	E	F
2-3	A-B	B	C	E	F
3-5	B	B-C	C	D	E
5-6	C	C-D	D	D	D
More than 6	C	D	D	D	D

Assuming that all activities involving spent fuel like handling and transportation takes place during day-time it is clear from above that wind categories to be encountered will be A, B, C and D. The effluent concentration has been calculated using equation 4.2 for 1, 10, 20 and 50 kilometers and an effective stack height of 89 metres (same as RAPP stack height) for SFSP. Stack height for the transportation accident has been taken as 2 metres.

4.3 COMPUTATION OF EXPOSURE AND DOSE:

Large exposure to radiation for a small time is called acute exposure. Continuous exposure to small doses is known as chronic exposure. The difference is brought out by our model. While the plume is airborne the 'cloud' would be dispersed and diluted by turbulent diffusion. When the cloud passed over an area, the people there will be exposed to the radiation emanating from the cloud. The exposure of people from the passage of the cloud is called acute exposure as it would be of relatively short duration and the radioactive material would be relatively concentrated. Acute exposure is mainly therefore external exposure to radioactive plume. In contrast, the exposure of people to the radioactive material deposited on the terrain is called chronic exposure. The length of exposure here is longer, and is mainly through radionuclide ingestion from milk, milk products, meat, vegetables and other foodstuff. Inhalation from resuspended radionuclides may be another path for chronic exposure.

4.3.1 External Exposure to Gama Rays:

The external exposure due to gamma rays is given by following equation [37]:

$$\dot{X} = 2.62 \times 10^5 \times \bar{E} \times 'CHI' \quad \text{R/Sec.}$$

where \bar{E} is the average energy of all of the gamma-rays per disintegration. If 'CHI' is given in Ci/m³ as is for our case then,

$$\dot{X} = 0.262 \times \bar{E} \times \text{'CHI'} \quad \text{R/Sec.} \quad (4.4)$$

The derivation of this equation is given in Appendix C. To obtain dose equivalent (rems) from exposure (roentgens), \dot{X} must be multiplied by the f-factor and quality factor. The f-factor is an energy dependent function which depends on the composition of the tissue. For γ -ray both the factors are approximately unity, so,

$$\dot{H} = 0.262 \times \bar{E} \times \text{'CHI'} \quad \text{rems/sec.} \quad (4.5)$$

4.3.2 External Dose From Plume - Beta Rays:

As given in Appendix C, the absorbed dose rate to air is,

$$D_{\text{air}} = 4.58 \times 10^5 \times \bar{E}_B \times \text{'CHI'} \quad \text{rad}$$

Since the skin is normally exposed to only one-half of the cloud,

$$\dot{D}_{\text{air}} = 2.29 \times 10^5 \times \bar{E}_B \times \text{'CHI'} \quad \text{rad} \quad (4.6)$$

In equation 4.5, 'CHI', is in Ci/cm³, when expressed in Ci/m³, \dot{D} becomes,

$$\dot{D}_{\text{air}} = 0.229 \times \bar{E}_B \times \text{'CHI'} \quad \text{rad} \quad (4.7)$$

CHAPTER V

RESULTS, CONCLUSIONS AND SUGGESTIONS FOR FURTHER WORK

5.1 RESULTS AND DISCUSSION:

The fission product inventory, and the decay of its activity with time is tabulated in Table 2.3. The behaviour of the fission products with time is shown by Fig. 2.1, and as expected, is found to be in conformity with established work [35]. Figure 2.1 shows that the behaviour of the fission products, specially due to the production of daughter isotopes, changes abruptly at three points, 100 hours, 120 days and 2 year. This helps to conclude the boundaries for the four groups as described in Section 2.3. The four groups mentioned are the very rapidly saturating, rapidly saturating, slowly saturating and very slowly saturating groups.

The Monte Carlo simulation was done with the aim to arrive at a distribution for the top event.

5.2 SAMPLE CODE USE ON SFSP TREE:

The output evaluation for sample size 1200 the distribution parameters were found to be

Mean = 0.02997 ,

Standard deviation = 0.01306

The dose equivalent rate in tissue is then given by,

$$\dot{H} = 0.229 \times \bar{E}_B \times 'CHI' \times f(d, E_{max}) \text{ rem} \quad (4.8)$$

where f is an experimentally determined function. The dose rate is largest at the surface of the skin, where $f = 1$, and decreases with distance (d) into the tissue. To be on the conservative side, we choose $f = 1$, so,

$$\dot{H} = 0.229 \times \bar{E}_B \times 'CHI' \quad (4.9)$$

The values of 'CHI' calculated as explained in Sec.4.2 are substituted in equations (4.5) and (4.9) together with \bar{E} and \bar{E}_B and the doses are calculated for the four wind condition A,B,C, and D at distances of 1 kilometre, 10 kilometre, 20 kilometre, 50 Kilometer. The total dose due to Gamma and Beta rays is tabulated in Appendix D.

4.4 FINAL RISK ESTIMATION:

Our objective throughout the study has been to estimate the risk to population due to accident involving spent fuel. The total exposure to population has been estimated in rems. The total risk has been estimated by using equation (1.1) and (1.2) as described in Chapter I.

So risk becomes,

$$\text{Risk to population} = \text{Probability of occurrence of accident} \times \text{release fraction} \times \text{Total dose exposure}$$

$$\text{or } R = P \times f \times \dot{H} \quad (4.10)$$

\dot{H} is estimated by using equation (4.5) and (4.9). The release fraction f estimation involves the knowledge and application of spent fuel can parameters, packing method, fraction mechanics, volatility and solubility of fission products, and impact analysis. Various experimental studies to determine the release fraction has been carried mostly in U.S., like dropping of cans from heights of 100 meters or more, simulation of impact on computer, actual accident analysis like impact between two trucks, impact against wall. But still reliable data about release fraction is yet to be established. We solved this problem by calculating the risk for 10, 20, 50 and 100 percent release. The risks are tabulated in Appendix D.

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The distribution confidence limits are as follows:

<u>Confidence (percent)</u>	<u>Function Value</u>
0.5	0.01081
1.0	0.01197
2.5	0.01287
5.0	0.01425
10.0	0.01637
20.0	0.01968
25.0	0.02118
30.0	0.0221
40.0	0.0243
50.0	0.027
60.0	0.02998
70.0	0.03396
75.0	0.03617
80.0	0.03861
90.0	0.04709
95.0	0.05411
97.0	0.07195
99.5	0.08799

The frequency distribution for the SFSP failure is plotted in Figure 5.1. To analyse the effect of trial run on the sample size on the distribution parameters, the sample size was varied as below:

<u>Sample Size</u>	<u>Mean</u>	<u>Standard Deviation</u>
400	0.030400	0.01343
600	0.030006	0.01298
1200	0.07997	0.01306
2200	0.02993	0.01286
3000	0.02992	0.01288

It is concluded from above that the sample size does not have much bearing on the distribution parameters.

5.3 SAMPLE CODE USE ON TYPICAL CANDU SPENT FUEL TRANSPORTATION ACCIDENT TREE:

The **SAMPLE** code was run for evaluating the distribution of the TOP event for this tree. The distribution parameters with the sample size is listed below:

<u>Sample Size</u>	<u>Mean</u>	<u>Standard Deviation</u>
500	0.2552×10^{-6}	0.4470×10^{-6}
1200	0.2374×10^{-6}	0.3793×10^{-6}
2000	0.2362×10^{-6}	0.3589×10^{-6}
3000	0.2423×10^{-6}	0.36300×10^{-6}

It is clear that though the best result were obtained with sample size 2000, the result was not meaningful. The variation of the distribution parameters due to variation of the error factors in the input were analysed with the

sample size being 2000. As seen from Table 3.2, units 1 and 3 are the critically important contributors through the units 2 and 11 are also important. The effect on the distribution parameters with variation of error factors is listed in Table 5.1.

Table 5.1: Effect of error factor on transportation tree distribution parameters.

Component 1	Component 2	Component 3	Component 4	Mean	Standard Deviation
10	10	10	10	0.2374×10^{-6}	0.3793×10^{-6}
20	10	20	10	0.4313×10^{-6}	0.1089×10^{-5}
20	10	20	20	0.4338×10^{-6}	0.1091×10^{-5}
20	10	10	10	0.3813×10^{-6}	0.9752×10^{-6}
10	10	20	10	0.2803×10^{-6}	0.6025×10^{-6}
30	10	20	10	0.5974×10^{-6}	0.1934×10^{-5}
30	10	30	10	0.6579×10^{-6}	0.2198×10^{-5}
5	10	5	10	0.1517×10^{-6}	0.1312×10^{-6}
10	10	5	10	0.2151×10^{-6}	0.3343×10^{-6}
20	10	5	10	0.3601×10^{-6}	0.966×10^{-6}
5	10	5	5	0.1487×10^{-6}	0.1304×10^{-6}

It is clear from Table 5.1, that the best result is obtained when the error factor of units 1 and 4 are 5 and the rest 9 units are 10. The conclusion drawn here is that the probability of components which are critically important

The transportation accident frequency distribution is shown in Fig. 5.2.

The point estimate of the transportation accident probability is 0.9249×10^{-7} where as using Smeeds formula (Eq. 3.14) we arrive at a figure,

$$\text{Probability of accident} = 4.68576 \times 10^{-8}$$

and using Jacobs formula (equation 3.15) we get,

$$\text{Probability of accident} = 4.3266 \times 10^{-8}$$

The point estimate is therefore 2.13 times the figure given by Jacobs formula which shows that the transportation accident fault tree gives a conservative value.

5.4 EVALUATED RISK RESULTS AND CONCLUSION:

The evaluation of risk to people due to accident involving spent fuel, which is the primary objective of the study has been evaluated and tabulated in Table 5.2 and Appendix D. The risk to people does not reduce exponentially with time but the behaviour is similar to the decay of fission product inventory as shown in Fig. 2.1. However, from Table 5.2, it can be concluded that the risk to people decays exponentially with distance. Also it is seen that the risk is minimum with wind condition A and progressively increases with Pasquill categories B, C and D. We can therefore conclude from this, that with unstable wind

contributors to the TOP event, have to be specified accurately. With these error factors and sample size 2000 being input parameters for the SAMPLE code, the output distribution parameters are:

$$\text{Mean} = 0.1489 \times 10^{-6} \quad \text{Standard devi.} = 0.1304 \times 10^{-6}$$

and the distribution confidence limits as given below:

<u>Confidence (Percent)</u>	<u>Function Value</u>
0.5	0.2351×10^{-7}
1.0	0.2612×10^{-7}
2.5	0.3331×10^{-7}
5.0	0.3939×10^{-7}
10.0	0.4905×10^{-7}
20.0	0.6336×10^{-7}
25.0	0.6928×10^{-7}
30.0	0.7715×10^{-7}
40.0	0.9284×10^{-7}
50.0	0.1109×10^{-6}
60.0	0.1321×10^{-6}
70.0	0.1593×10^{-6}
75.0	0.1991×10^{-6}
80.0	0.2032×10^{-6}
90.0	0.2838×10^{-6}
95.0	0.4015×10^{-6}
97.0	0.4969×10^{-6}
99.0	0.6890×10^{-6}
99.5	0.7939×10^{-6}

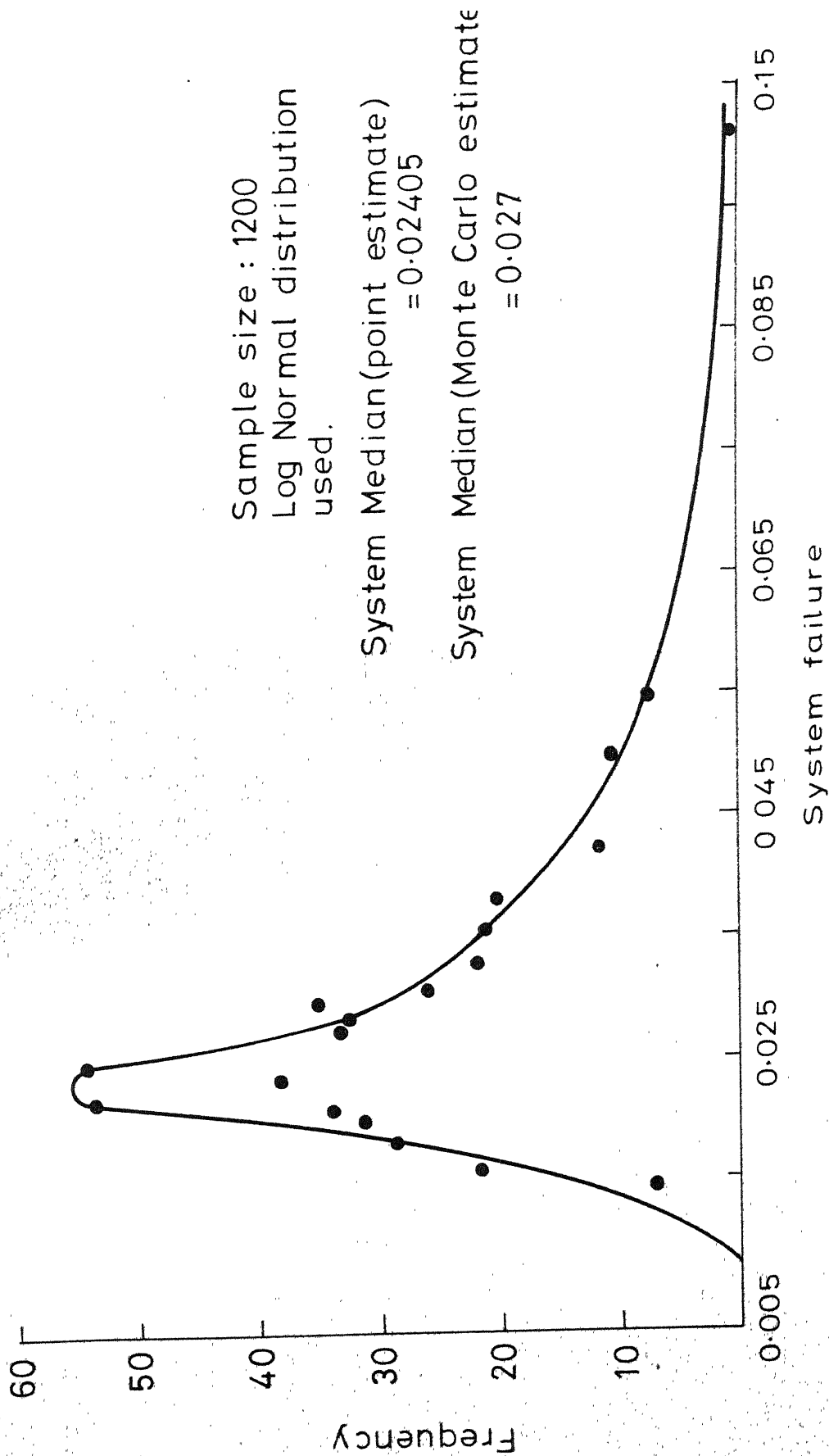


Fig. 5.1 SFSP failure frequency distribution

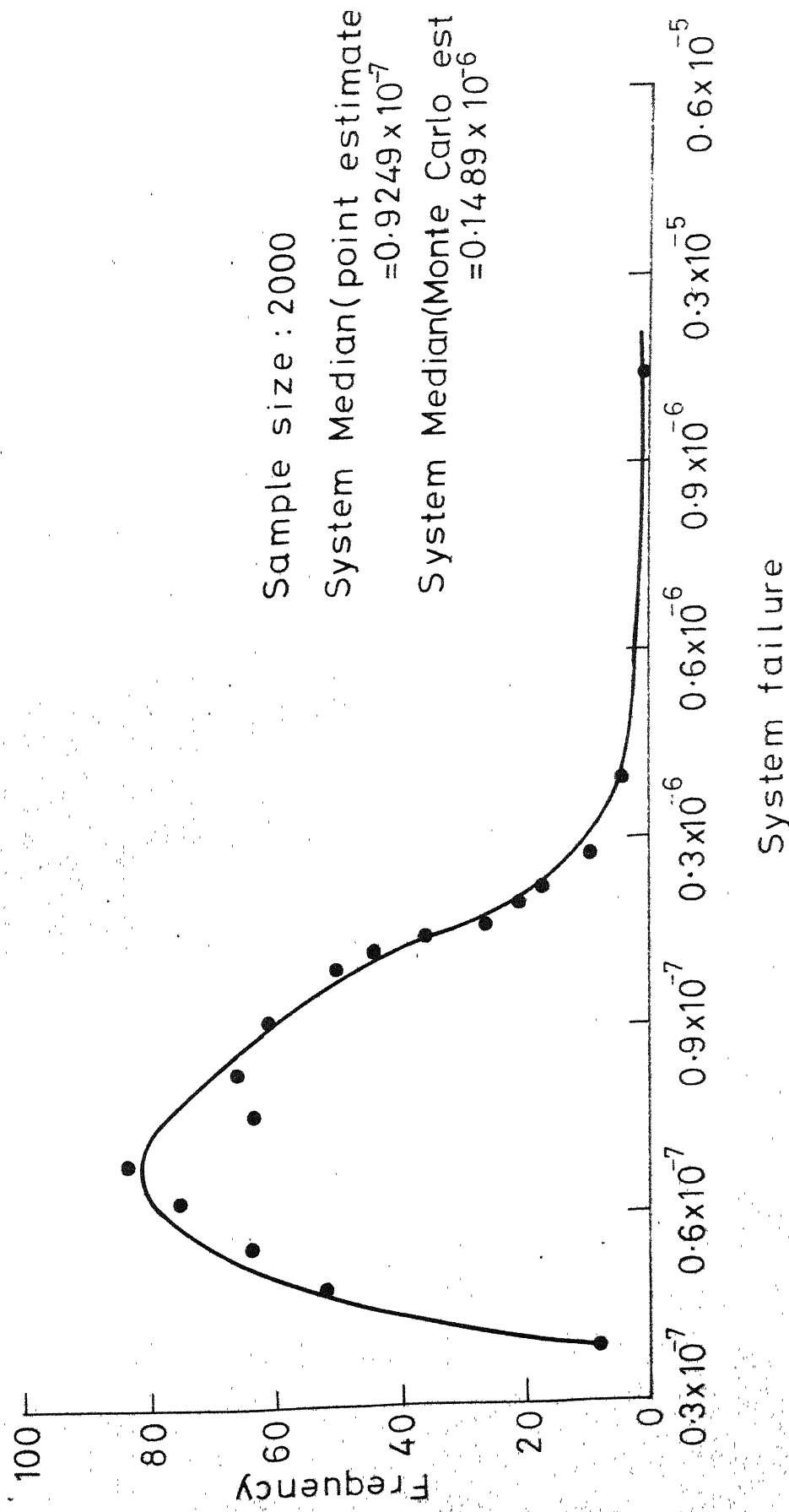


Fig. 5.2 Frequency distribution of typical CANDU spent fuel transportation accident tree

condition the risk to people is considerably less and increases as the wind condition becomes more stable. The risk to population during transportation of the spent fuel is tabulated in Table 5.3. It is obvious that the magnitude of the risk is very small and therefore, transportation of spent fuel by truck is quite safe. It can also be seen, that the risk to people during transportation of the spent fuel by rail (accident probability = 0.14×10^{-7} , probability of radioactivity release = 0.01) is comparable to the risk during transportation by truck.

5.5 SUGGESTIONS FOR FUTURE WORK:

The present work can be extended to include:

- 1) The list of fission products can be increased to make it more comprehensive. The number of fission products considered are 79 but there is scope for inclusion of other fission products.
- 2) The number of decay chain considered in this work was three. But for a better representation of the decay of fission product, the number of decay chain should be at least 5 or 6.
- 3) Sabotage is an important component whose inclusion in the fault trees of both SFSP and transportation accident, can make the risk analysis more realistic. Usually sabotage is neglected due to lack of adequate data base but the importance is being recognised now and frequent suggestion for its inclusion are coming up.

4) The effect of terrain on effluent dispersion is a major area, where more research effort has to be put. This involves more research in the means of geography, geology, demography and meteorology.

5) The present work can be extended to include the dose due to inhalation of radio nuclides and ingestion through foodstuff.

Table 5.2

Risk to people following accident in SFSP.
Distance from reactor site = 10 km.

Time after discharge	Weather condition			
	A	B	C	D
1 hour	0.222×10^{-1}	0.2735	0.5584×10^{-1}	0.2168×10^{-2}
5 hour	0.3594×10^{-2}	0.4429×10^{-1}	0.90522	0.3512×10^{-1}
10 hour	0.1307×10^{-2}	0.1609×10^{-1}	0.32893	0.1276×10^{-1}
25 hour	0.2531×10^{-3}	0.3114×10^{-2}	0.6365×10^{-1}	0.1234
50 hour	0.644191×10^{-4}	0.7927×10^{-3}	0.162×10^{-1}	0.6286×10^{-1}
100 hour	0.2085×10^{-4}	0.2582×10^{-3}	0.524×10^{-2}	0.2033×10^{-1}
10 days	0.0350×10^{-5}	0.1143×10^{-3}	0.234×10^{-2}	0.909×10^{-2}
25 days	0.569×10^{-5}	0.700×10^{-4}	0.143×10^{-2}	0.555×10^{-2}
50 days	0.135×10^{-5}	0.3131×10^{-4}	0.8443×10^{-3}	0.927×10^{-3}
120 days	0.9488×10^{-6}	0.1167×10^{-4}	0.2386×10^{-3}	0.125×10^{-3}
280 days	0.8300×10^{-7}	0.108×10^{-5}	0.2084×10^{-4}	0.809×10^{-4}
400 days	0.194×10^{-7}	0.362×10^{-6}	0.741×10^{-5}	0.181×10^{-4}
Distance from Reactor Site = 50 km.				
1 hour	0.9877×10^{-3}	0.395×10^{-2}	0.2236xx	0.1582×10^{-1}
5 hour	0.1549×10^{-4}	0.6399×10^{-3}	0.3622×10^{-1}	0.2563
10 hour	0.5813×10^{-4}	0.232×10^{-3}	0.131×10^{-1}	0.931×10^{-1}
25 hour	0.1124×10^{-4}	0.4499×10^{-4}	0.2547×10^{-2}	0.1802×10^{-1}
50 hour	0.2866×10^{-5}	0.114×10^{-4}	0.648×10^{-3}	0.458×10^{-2}
100 hour	0.926×10^{-6}	0.370×10^{-5}	0.209×10^{-3}	0.148×10^{-2}
20 days	0.414×10^{-6}	0.165×10^{-5}	0.938×10^{-4}	0.664×10^{-3}
25 days	0.252×10^{-6}	0.93×10^{-6}	0.572×10^{-4}	0.405×10^{-3}
50 days	0.149×10^{-7}	0.596×10^{-6}	0.337×10^{-5}	0.239×10^{-4}
120 days	0.913×10^{-8}	0.113×10^{-6}	0.913×10^{-6}	0.873×10^{-5}
280 days	0.368×10^{-8}	0.147×10^{-7}	0.335×10^{-6}	0.391×10^{-5}
400 days	0.981×10^{-9}	0.524×10^{-8}	0.996×10^{-9}	0.812×10^{-6}

Table 5.3

Risk due to transportation of spent fuel by road freight.

	Weather condition			
	A	B	C C	D
Risk at a distance from accident site				
100 metres	9.082×10^{-10}	1.677×10^{-9}	3.082×10^{-9}	6.886×10^{-9}
200 metres	1.101×10^{-10}	2.221×10^{-10}	3.844×10^{-10}	1.006×10^{-9}
500 metres	1.232×10^{-11}	3.539×10^{-11}	7.309×10^{-11}	2.115×10^{-10}
1 Kilometre	7.55×10^{-13}	3.906×10^{-12}	1.087×10^{-11}	3.36×10^{-11}
2 Kilometre	3.636×10^{-15}	1.907×10^{-14}	6.537×10^{-14}	1.313×10^{-14}

Risk due to transportation of spent fuel by rail freight.

100 metre	1.146×10^{-9}	2.117×10^{-9}	3.89×10^{-9}	8.692×10^{-9}
200 metre	1.39×10^{-10}	2.803×10^{-10}	4.852×10^{-10}	1.269×10^{-9}
500 metre	1.555×10^{-11}	4.467×10^{-11}	9.226×10^{-11}	2.669×10^{-10}
1 Kilometre	9.531×10^{-13}	4.93×10^{-12}	1.372×10^{-11}	4.241×10^{-11}
2 Kilometre	4.59×10^{-15}	2.407×10^{-14}	8.252×10^{-14}	1.657×10^{-14}

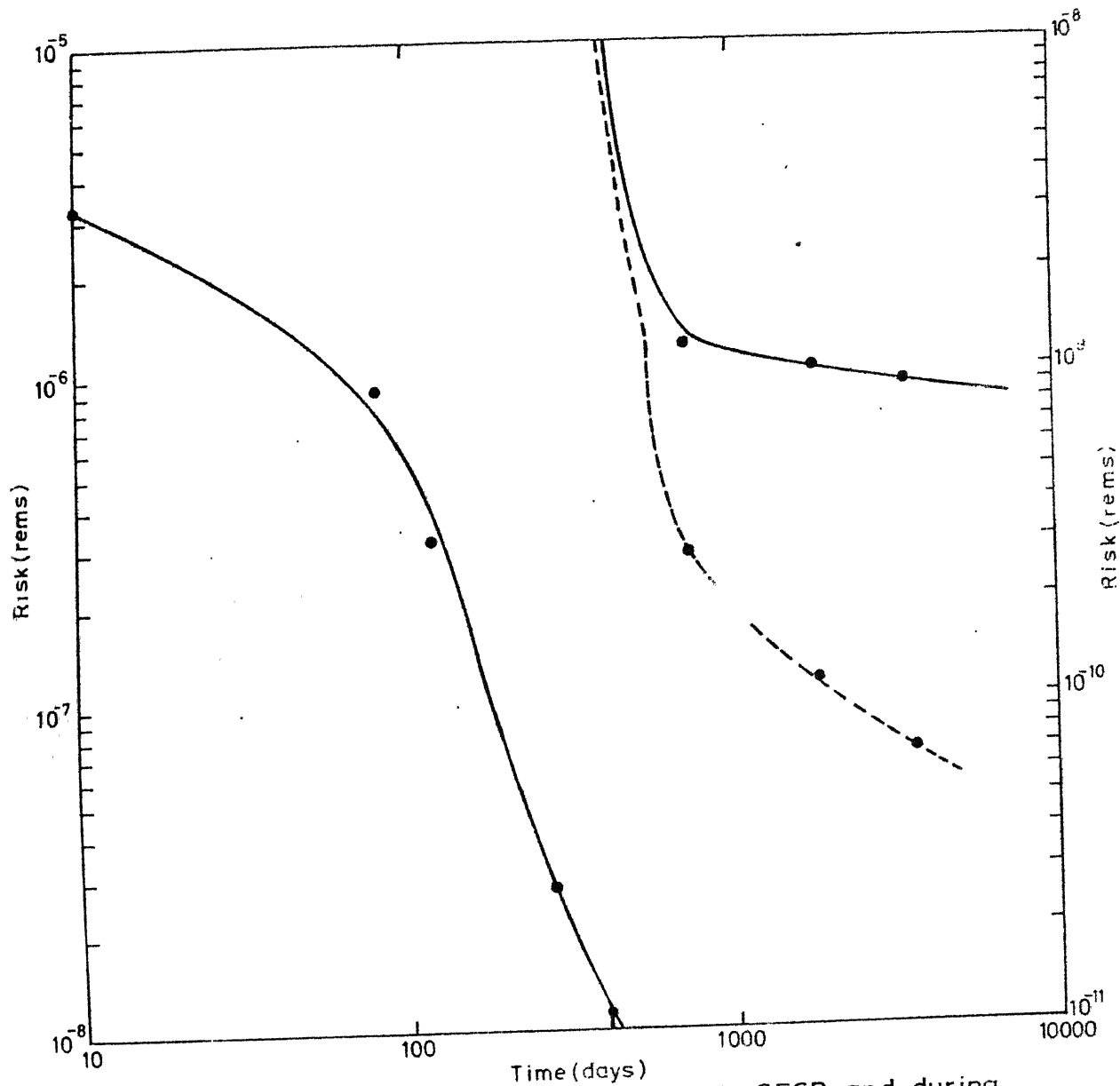


Fig. 5-3 Risk to population due to accident in SFSP and during transportation of spare fuel

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APPENDIX A

FISSION PRODUCT COMPOSITION

F.P.Name	Yield from U ²³⁵ (Percent)	Yield from Pu ²³⁹ (percent)	Daughter Name
Se ⁸⁷	2		Br ⁸⁷
Se ⁸⁵	1.1		Br ⁸⁵
I ¹³⁶	3.1	2.1	Xe ¹³⁶
Sn ¹³⁰	2		Te ¹³⁰
Br ⁸⁴	0.019		Kr ⁸⁴
Ga ⁷⁴	0.00035		Ge ⁷⁴
As ⁷⁹	0.056		Se ⁷⁹
Se ⁸¹	0.14		Br ⁸¹
Rh ¹⁰⁷	0.19		Ag ¹⁰⁷
Sm ¹⁵⁵	0.033		Eu ¹⁵⁵
Se ⁸³	0.22		Br ⁸³
I ¹²⁸	0.00003		Xe ¹²⁸
Br ⁸⁴	0.92		Kr ⁸⁴
Mb ⁹⁸	0.064	0.2	Mo ⁹⁸
I ¹³⁴	7.8		Xe ¹³⁴
Sn ¹²⁸	0.37		Sb ¹²⁸
Se ⁸¹	0.0084		Br ⁸¹
Eu ¹⁵⁸	0.002		Gd ¹⁵⁸
Ba ¹³⁹	6.55	5.87	Ca ¹³⁹
As ⁷⁸	0.02		Se ⁷⁸
Ge ⁷⁸	0.02		As ⁷⁸
Br ⁸³	0.51	0.084	Kr ⁸³
Sr ⁹²	5.3		Y ⁹²
Cd ¹¹⁷	0.011		Sn ¹¹⁷
La ¹⁴¹	6.4	5.7	Ce ¹⁴¹
Ru ¹⁰⁵	0.9		Rh ¹⁰⁵
Ga ⁷³	0.00011		Ge ⁷³
I ¹³⁵	6.1	5.7	Xe ¹³⁵

contd...

F.P.Name	Yield from U ²³⁵ (Percent)	Yield from Pu ²³⁹ (Percent)	Daughter Name
Xe ¹³⁵	6.3		Cs ¹³⁵
Sr ⁹¹	5.81	2.43	Y ⁹¹
Y ⁹³	6.1	Zr	Zr ⁹³
Ge ⁷⁷	0.0031		As ⁷⁷
I ¹³⁰	0.005		Xe ¹³⁰
Pd ¹⁰⁹	0.03	1.4	Ag ¹⁰⁹
Eu ¹⁵⁷	0.0078		Gd ¹⁵⁷
Zr ⁹⁷	5.9	5.5	Nb ⁹⁷
Gd ¹⁵⁹	0.00107		Eu ¹⁵⁹
I ¹³³	6.6		Cs ¹³³
Pd ¹¹²	0.01	0.120	Cd ¹¹²
Nb ⁹⁶	0.00061	0.0036	Mo ⁹⁶
Sn ¹²¹	0.015	0.043	Sb ¹²¹
Ce ¹⁴³	5.7	5.3	Pr ¹⁴³
Br ⁸²	0.00004		Kr ⁸²
Rh ¹⁰⁵	0	3.9	Ag ¹⁰⁵
As ⁷⁷	0.0083		Se ⁷⁷
Sm ¹⁵³	0.15	0.37	Eu ¹⁵³
Zn ⁷²	0.00016	0.00012	Ga ⁷²
Cd ¹¹⁵	0.0097	0.0038	In ¹¹⁵
Mo ⁹⁹	6.06	6.1	Tc ⁹⁹
Te ¹³²	4.7	5.1	Xe ¹³²
Dy ¹⁶⁶	0	0.00006	Er ¹⁶⁶
Sb ¹²⁷	0.13	0.39	I ¹²⁷
Xe ¹³³	6.62	6.69	Cs ¹³³
Tb ¹⁶¹	0.00007	0.0039	Dy ¹⁶¹
Ag ¹¹¹	0.019	0.23	Cd ¹¹¹
I ¹³¹	3.1	3.77	Xe ¹³¹
Sn ¹²⁵	0.0013	0.071	Sb ¹²⁵
Nd ¹⁴⁷	2.7	2.2	Pm ¹⁴⁷

Contd...

F.P.Name	Yield from U ²³⁵ (Percent)	Yield from Pu ²³⁹ (Percent)	Daughter Name
Ba ¹⁴⁰	6.35	5.4	Ce ¹⁴⁰
Cs ¹³⁶	0.0068	0.11	Ba ¹³⁶
Eu ¹⁵⁶	0.014	0.11	Gd ¹⁵⁷
Rb ⁸⁶	0.00002	0.00002	Br ⁸⁶
Ce ¹⁴¹	6.0	5.1	Pr ¹⁴¹
Te ¹²⁹	0.35		I ¹²⁹
Ru ¹⁰³	3.0	5.67	Rh ¹⁰³
Cd ¹¹⁵	0.0001	0.003	In ¹¹⁵
Sr ⁸⁹	4.79	1.71	Y ⁸⁹
Zr ⁹⁵	6.20	5.8	Nb ⁹⁵
Te ¹²⁷	0.035		I ¹²⁷
Sn ¹²³	0.0013		Sb ¹²³
Ce ¹⁴⁴	6.0	3.79	Nd ¹⁴⁴
Ru ¹⁰⁶	0.38	4.57	Pd ¹⁰⁴
Sb ¹²⁵	0.021		Te ¹²⁵
Pm ¹⁴⁷	0	1.94	Nd ¹⁴⁷
Eu ¹⁵⁵	0.033		Gd ¹⁵⁵
Kr ⁸⁵	0.293	0.127	Rb ⁸⁵
Sr ⁹⁰	5.77	2.25	Y ⁹⁰
Cs ¹³⁷	6.15	6.63	Ba ¹³⁷
Sm ¹⁵¹	4.4	0.8	Eu ¹⁵¹

APPENDIX B

B.1 TOP EVENT AND BOOLEAN ALGEBRA:

The TOP event which describes the failure of the system, can be expressed as a Boolean algebra function of primary events. The following basic properties and laws of Boolean algebra are generally used to simplify the TOP event function and to eliminate redundant event

a. Identities:

$$1. \quad A + A = A$$

$$2. \quad A \cdot A = A$$

b. Distributive Laws:

$$1. \quad A(B + C) = (A \cdot B) + (A \cdot C)$$

$$2. \quad A + (B \cdot C) = (A + B) \cdot (A + C)$$

c. Laws of Absorption:

$$1. \quad A + (A \cdot B) = A$$

$$2. \quad A (A + B) = A$$

Once the TOP event is expressed as a simplified Boolean function of primary events, the probability of the former can be related to that of later by use of basic operational laws to combine probability are given below:

a. UNION:

Boolean Expression : $T = A + B$

Probability expression

$$P(T) = P(A) + P(B) - P(A.B)$$

b. INTERSECTION:

Boolean expression : $T = A.B$

Probability expression,

$P(T) = P(A).P(B/A)$, if A and B are dependent

or $P(T) = P(A).P(B)$, if A and B are independent.

The rare event approximation is applicable when the intersection probability $P(A.B)$ is much smaller, than the individual probabilities, $P(A)$ and $P(B)$. If A and B are independent than $P(A.B) = P(A).P(B)$ and the rare event approximation will be valid if cross product from $P(A).P(B)$ is much smaller than $P(A)$ and $P(B)$. This will be the case when $P(A)$ and $P(B)$ are less than approximately 0.1 [7]. Whether the events are independent or dependent and regardless of probabilities, the rare event approximation will always give a conservative estimate [5], whether the fault trees are simple or complex, the essential idea is to obtain all the minimal cut sets of the TOP event such that the TOP event can be written as

$$T = M_1 + M_2 + \dots M_n \quad (B.1)$$

where M_i , the minimal cut-set also termed as critical paths, is intersection of primary events C_{ik} ,

$$M_i = C_{i1} C_{i2} C_{i3} \dots C_{im} \quad (B.2)$$

and no M_i is a subset of another M_j . Applying probability laws given above to TOP event expression of equation (B.1) we get,

$$P(T) = \sum_{i=1}^n P(M_i) - \sum_{j=1}^n \sum_{j=1}^n P(M_i) P(M_j) \quad (i \neq j)$$

+ Probability of all possible triple combinations

If rare event approximation is applied then,

$$P(T) \approx \sum_{i=1}^n P(M_i) \quad (B.3)$$

If primary events are independent,

$$P(M_i) = \sum_{k=1}^m P(C_{ik}) \quad (B.4)$$

B.2 JUSTIFICATION OF THE CHOICE OF LOG-NORMAL DISTRIBUTION:

The log-normal is obtained by a transformation of a normally distributed variable. It is often used [7] in modelling system when factors or percentages characterise the input component data variation. If X is a random variable, which can vary between X_0/f and $X_0.f$, where X_0 is some

mid point reference value and f some factor, then a log-normal fits appreciably.

In this study, the LOG NORMAL is suitable because the components and the input data in general does vary by factors. For example, a potential damage to the SFSP pool due to earth-quake, can have a probability of occurrence between 4×10^{-6} to 2×10^{-8} per year. Besides giving adequate fitting to cases evaluated [7], the log-normal is applicable here from reliability and statistical considerations. They are:

- 1) The positive skewness in the log-normal distribution form, can incorporate general reliability associated behaviour of the data. The positive skewness accounts for the occurrence of less likely but large deviation.
- 2) As an a priori distribution the log-normal gives coverage to errors which can be skewed towards large values. In general, the average value is greater than the most probable value, which provides a protective, positive type bias, which is retained when the distribution is propagated, say as for the calculation of the top event in the fault tree.
- 3) The log normal distribution being a very flexible one, can under the applicable situation, adapt itself as a near normal type shape or a near exponential type shape.

4) The log-normal has an established history of useful representation when relative variations (factors) characterise the random variable. Its application as a general distribution for modelling physical and reliability process is established and has been validated.

5) The assessed data comprises of reliability data in the form of probabilities. If the probabilities are decomposed into products of probabilities representing requisites for failure, then when the central-limit theorem is applied the log normal is the resulting distribution.

B.3 DEPENDENCY TREATMENT:

The handling of dependencies and common mode failures while quantifying them is discussed here. As defined earlier, a common mode failure is a failure combination (two or more failures), which is dependent, or which has more basic common causes. In SFSP fault tree the common mode failure events are components 1 and 6, 2 and 5 and 7 and 10. The following procedure is used to determine the additional common-mode contribution of X_{CM} .

Taking the case of component 1 and 6, the contribution X_{CM} is assessed to be the square root of the product of the upper and lower bounds [7, 28] as defined by $X_{upper} = X_1 = X_6$ and $X_{lower} = X_1 \cdot X_6$ where X_1 and X_6 are the values of $X(1)$ and $X(6)$. The contribution is $\sqrt{X_{upper} X_{lower}}$ and hence

$$X_{CM1} = \sqrt{X_1 \cdot X_1 \cdot X_6}$$

$$X_{CM2} = \sqrt{X_2 \cdot X_2 \cdot X_5}$$

$$X_{CM3} = \sqrt{X_7 \cdot X_7 \cdot X_{10}}$$

Since the fault trees include both-single component and common-mode components, it is obvious, that the TOP event is dominated by the higher probability, single failure contributions, and the common mode components are second order contributors.

APPENDIX C

C.1 EFFLUENT DIFFUSION:

The diffusion of pollutants is chiefly due to turbulent eddies in the atmosphere. The individual particles become increasingly separated from one another as the result of local atmospheric turbulence and temperature variation. The process is called turbulent diffusion.

Let 'CHI' be the concentration of some effluent as a function of space and time. Then 'CHI' is determined by the time - dependent diffusion equation,

$$K \nabla^2 [\text{CHI}] = \frac{\partial [\text{CHI}]}{\partial t} \quad (\text{C.1})$$

where K is the diffusion coefficient (cm²/sec.). Consider a pointsource located at the origin of coordinates which emits at time t = 0 an instantaneous isotropic puff containing a total of Q units of effluent into an infinite stationary atmosphere. It is shown in pp. 192 [36], that the solution to equation (C.1) is

$$\text{CHI} (r, t) = \frac{Q}{(4 \pi Kt)^{3/2}} e^{-r^2/4 Kt} \quad (\text{C.2})$$

where r is the distance from the origin C.2 is based on the assumption that the atmosphere is isotropic. When the atmosphere is non-isotropic, the solution is

$$\begin{aligned} \text{'CHI' } (x,y,z,t) &= \frac{Q}{(4t)^{3/2} (K_x K_y K_z)^{1/2}} \\ &\times \exp \left[-\frac{1}{4t} \left(\frac{x^2}{K_x} + \frac{y^2}{K_y} + \frac{z^2}{K_z} \right) \right] \quad (C.3) \end{aligned}$$

The movement of effluent in the direction of the wind is due to the wind itself and not due to diffusion. So the diffusion in x-direction is ignored. The physical picture is that of a thin disk which is moving along the x-axis with wind speed \bar{v} , which is continuously spreading out in y and z direction as it moves. So equation (C.3) is reduced to

$$\begin{aligned} \text{'CHI' } (x,y,z,t) &= \frac{Q}{\pi \bar{v} t (K_y K_z)^{1/2}} \\ &\times \exp \left[-\frac{1}{4t} \left(\frac{y^2}{K_y} + \frac{z^2}{K_z} \right) \right] \quad (C.4) \end{aligned}$$

It can be seen that the effluent is Gaussian in the y- and z-direction, with the standard deviations σ_y and σ_z given by,

$$2\sigma_y^2 = 4tK_y ; \quad 2\sigma_z^2 = 4tK_z$$

So,

$$'CHI' (x,y,z) = \frac{Q}{2 \pi \bar{v} \sigma_y \sigma_z} \exp \left[-\left(\frac{y^2}{2\sigma_y^2} + \frac{z^2}{2\sigma_z^2} \right) \right] \quad (C.5)$$

It was assumed till here that the effluents are omitted at the origin of coordinates. To consider emission at an altitude h , the method of images was applied. So for the concentration at ground level, $z = 0$

$$'CHI' = \frac{Q}{\pi \bar{v} \sigma_y \sigma_z} \exp \left[-\left(\frac{y^2}{2\sigma_y^2} + \frac{h^2}{2\sigma_z^2} \right) \right] \quad (C.6)$$

The value of 'CHI' is largest along the centre line of the plume where $y = 0$. The concentration there is,

$$'CHI' = \frac{Q}{\pi \bar{v} \sigma_y \sigma_z} \exp \left(-\frac{h^2}{2\sigma_z^2} \right) \quad (C.7)$$

C.2 EXTERNAL DOSE FROM PLUME:

C.2.1 Gama-Rays:

To compute the Gama-ray exposure from a specified radiation field, it is necessary to determine the energy absorbed from the Gama-rays in the air, and then convert it to roentgens, where an exposure of 1R corresponds to an energy deposition of

$$1R = 5.47 \times 10^7 \text{ MeV/g} \quad (C.8)$$

Since $1 \text{ MeV} = 1.60 \times 10^{-6} \text{ ergs}$, it follows that

$$\begin{aligned} 1 \text{ R} &= 5.47 \times 10^7 \times 1.60 \times 10^{-6} \\ &= 87.5 \text{ ergs/g.} \end{aligned}$$

The total collision density at a point where Gamma-ray intensity is I is given by,

$$F = I \mu$$

where μ is the total attenuation coefficient. If the gamma-ray was absorbed at each collision, then the rate at which energy is deposited per unit volume in the medium will be $EF = EI \mu$, where E is the energy of gamma-rays.

With both the photo electric effect and pair production, the incident photon is absorbed and most secondary radiation - the X-rays, electrons and annihilation radiation is also absorbed in the medium. In Compton scattering, the only energy deposited is the kinetic energy of the recoiling electron. The energy deposition rate per unit volume by Compton scattering is $E I \mu_{ca}$ where μ_{ca} is the Compton absorption coefficient. So the total energy deposition rate W per unit volume can be written as

$$\begin{aligned} W &= E I (\mu_{pe} + \mu_{pp} + \mu_{ca}) \\ &= E I \mu_a \end{aligned} \tag{C.9}$$

where $\mu_a = \mu_{pe} + \mu_{pp} + \mu_{ca}$ is called the energy absorption coefficient. So it follows from equation (C.8) and (C.9) that the exposure rate \dot{X} is, (where P denotes density of air)

$$\begin{aligned}\dot{X} &= I E (\mu_a/P)^{\text{air}} / 5.47 \times 10^7 \\ &= 1.83 \times 10^{-8} I E (\mu_a/P)^{\text{air}} \quad \text{R/Sec.}\end{aligned}$$

If the gamma-ray buildup flux is denoted by $\phi_{\gamma b}$ then the above equation can be rewritten as

$$\dot{X} = 1.83 \times 10^{-8} \phi_{\gamma b} E(-\mu_a/P)^{\text{air}} \quad \text{R/Sec.} \quad (\text{C.10})$$

It has been shown in [36] that the build up flux at the centre of the cloud is

$$\phi_{\gamma b} = \frac{S}{2\mu_a}$$

where S is equal to 3.7×10^{10} 'CHI' where the radionuclide concentration in the cloud is 'CHI' Ci/cm³.

Therefore equation (C.10) becomes,

$$\begin{aligned}\dot{X} &= 1.83 \times 10^{-8} \frac{3.7 \times 10^{10} \times \text{'CHI'}}{2\mu_a} E\left(\frac{\mu_a}{P}\right)^{\text{air}} \\ &= 2.62 \times 10^5 \times \text{'CHI'} \times E \quad \text{R/Sec}\end{aligned}$$

where $P = 1.293 \times 10^{-3}$ g/cm³ is the density of air. If more than one gamma-ray is emitted by the nuclide

$$\dot{X} = 2.62 \times 10^5 \times \text{'CHI'} \times \bar{E} \quad \text{R/Sec.}$$

If 'CHI' is given in Ci/m³ then,

$$\dot{X} = 0.262 \times \text{'CHI'} \times \bar{E} \quad (\text{C.11})$$

The dose equivalent can be found by multiplying above equation by f-factor and quality factor. However, since both the factors are approximately unity,

$$\dot{H} = 0.262 \times \text{'CHI'} \times \bar{E}$$

C.2.2 Beta-Rays:

At any point in an infinite cloud, as many beta-rays are absorbed per cm³/sec. as are emitted. If the cloud contains a radionuclide of 'CHI' Ci/cm³ emitting an average of \bar{E}_B MeV per disintegration, the total amount of energy absorbed is $3.7 \times 10^{10} \times \text{'CHI'} \times \bar{E}_B \times 1.6 \times 10^{-6}$ ergs/cm³-sec. This energy is absorbed in air having a mass of 1.293×10^{-3} g/cm³, so that the energy absorption rate is

$$\frac{3.7 \times 10^{10} \times \text{'CHI'} \times \bar{E}_B \times 1.6 \times 10^{-6}}{1.293 \times 10^{-3}}$$

$$= 4.58 \times 10^7 \times \text{'CHI'} \times \bar{E}_B \text{ erg/g-sec.}$$

The absorption of 100 erg/g gives an absorbed dose of 1 rad. Thus the absorbed dose rate to air,

$$\dot{D}_{\text{air}} = 4.58 \times 10^5 \times \text{'CHI'} \times \bar{E}_B \text{ rad/sec.}$$

Since the skin is exposed to only one-half of a cloud,

$$\dot{D}_{\text{air}} = 2.29 \times 10^5 \times \text{'CHI'} \times \bar{E}_B \text{ rad/sec.}$$

where 'CHI' is expressed in Ci/m³, \dot{D}_{air} becomes,

$$\dot{D}_{\text{air}} = 0.229 \times \text{'CHI'} \times \bar{E}_B$$

The dose equivalent rate in tissue is then given by,

$$\dot{H} = 0.229 \times \text{'CHI'} \times \bar{E}_B \times f$$

where f is an experimentally determined function. The dose rate is largest at the surface of the skin where $f = 1$, and decreases rapidly inside the tissue. To be conservative, therefore, skin dose is computed with $f = 1$.

DOSE AND RISK TO PEOPLE AT A DISTANCE OF 1 KILOMETRE
FROM SFSP

Time after discharge	Weather Condition			
	A	B	C	D
1 hour				
Dose	0.137×10^4	0.625×10^4	0.865×10^4	0.141×10^5
50% risk	0.185×10^2	0.844×10^2	0.116×10^3	0.190×10^3
20% risk	0.743×10^1	0.337×10^2	0.467×10^2	0.763×10^2
5 hour				
Dose	0.222×10^3	0.101×10^4	0.140×10^4	0.229×10^4
50% risk	0.301×10^1	0.137×10^2	0.189×10^2	0.309×10^2
20% risk	0.120×10^1	0.546×10^1	0.757×10^1	0.124×10^2
10 hour				
Dose	0.810×10^2	0.367×10^3	0.509×10^3	0.832×10^3
50% risk	0.109×10^1	0.496×10^1	0.687×10^1	0.112×10^2
20% risk	0.4375	0.198×10^1	0.275×10^1	0.449×10^1
25 hour				
Dose	0.156×10^2	0.712×10^2	0.985×10^2	0.161×10^3
50% risk	0.2116	0.9612	0.133×10^1	0.217×10^1
20% risk	0.847×10^{-1}	0.3844	0.5323	0.8498
50 hour				
Dose	0.399×10^1	0.181×10^2	0.250×10^2	0.409×10^2
50% risk	0.538×10^{-1}	0.2446	0.3387	0.5534×10^{-1}
20% risk	0.2154×10^{-1}	0.978×10^{-1}	0.1354	0.221×10^{-1}
100 hour				
Dose	0.129×10^1	0.586×10^1	0.8116×10^1	0.1326×10^2
50% risk	0.174×10^{-1}	0.791×10^{-1}	0.1095	0.1791
20% risk	0.697×10^{-2}	0.316×10^{-1}	0.0438	0.716
10 days				
Dose	0.5776	0.262×10^1	0.363×10^1	0.5934×10^1
50% risk	0.779×10^{-2}	0.354×10^{-1}	0.490×10^{-1}	0.801×10^{-1}
20% risk	0.311×10^{-2}	0.141×10^{-1}	0.190×10^{-1}	0.320×10^{-1}

contd...

Time after discharge	Weather Condition			
	A	B	C	D
25 days				
Dose	0.3524	0.160×10^{-1}	0.221×10^{-1}	0.362×10^{-1}
50% risk	0.475×10^{-2}	0.216×10^{-1}	0.299×10^{-1}	0.488×10^{-1}
20% risk	0.190×10^{-2}	0.864×10^{-2}	0.119×10^{-1}	0.195×10^{-1}
50 days				
Dose	0.907×10^{-1}	0.9443	0.9307	0.9136
50% risk	0.280×10^{-3}	0.127×10^{-2}	0.676×10^{-2}	0.9884×10^{-2}
20% risk	0.112×10^{-3}	0.510×10^{-3}	0.346×10^{-2}	0.496×10^{-2}
120 days				
Dose	0.587×10^{-1}	0.2669	0.3696	0.6039
50% risk	0.293×10^{-3}	0.95×10^{-3}	0.498×10^{-2}	0.8152×10^{-2}
20% risk	0.149×10^{-3}	0.42×10^{-3}	0.199×10^{-2}	0.293×10^{-2}
280 days				
Dose	0.514×10^{-2}	0.233×10^{-1}	0.323×10^{-1}	0.528×10^{-1}
50% risk	0.694×10^{-4}	0.315×10^{-3}	0.436×10^{-3}	0.713×10^{-3}
20% risk	0.277×10^{-4}	0.126×10^{-3}	0.174×10^{-3}	0.285×10^{-3}
400 days				
Dose	0.182×10^{-2}	0.8296×10^{-2}	0.114×10^{-1}	0.1877×10^{-1}
50% risk	0.246×10^{-4}	0.112×10^{-3}	0.155×10^{-3}	0.253×10^{-3}
20% risk	0.986×10^{-5}	0.448×10^{-4}	0.820×10^{-4}	0.101×10^{-3}

Risk to people at a distance of 20 Km from SFSP

1 hour				
Dose	0.2926	0.395×10^{-1}	0.7465×10^{-2}	0.357×10^{-3}
50% risk	0.395×10^{-2}	0.533×10^{-1}	0.1007×10^{-1}	0.483×10^{-1}
20% risk	0.158×10^{-2}	0.2133×10^{-1}	0.4031	0.193×10^{-1}
5 hour				
Dose	0.474×10^{-1}	0.6398	0.120×10^{-2}	0.579×10^{-2}
50% risk	0.639×10^{-3}	0.863×10^{-2}	0.1632	0.7824
20% risk	0.255×10^{-3}	0.345×10^{-2}	0.652×10^{-1}	0.3129
10 hour				
Dose	0.172×10^{-1}	0.2325	0.439×10^{-1}	0.290×10^{-2}
50% risk	0.232×10^{-3}	0.313×10^{-2}	0.593×10^{-1}	0.2843
20% risk	0.930×10^{-4}	0.1255×10^{-2}	0.237×10^{-1}	0.1137

contd...

Time after discharge	Weather Condition			
	A	B	C	D
25 hour				
Dose	0.333×10^{-2}	0.449×10^{-1}	0.8501	0.4075×10^1
50% risk	0.449×10^{-4}	0.607×10^{-3}	0.1147×10^{-1}	0.5502×10^{-1}
20% risk	0.179×10^{-4}	0.242×10^{-3}	0.459×10^{-2}	0.220×10^{-1}
50 hour				
Dose	0.848×10^{-3}	0.114×10^{-1}	0.2163	0.103×10^1
50% risk	0.114×10^{-4}	0.154×10^{-3}	0.292×10^{-2}	0.140×10^{-1}
20% risk	0.458×10^{-5}	0.618×10^{-4}	0.116×10^{-2}	0.560×10^{-2}
100 hour				
Dose	0.274×10^{-3}	0.37×10^{-2}	0.699×10^{-1}	0.3355
50% risk	0.370×10^{-5}	0.50×10^{-4}	0.944×10^{-3}	0.457×10^{-2}
20% risk	0.148×10^{-5}	0.2×10^{-4}	0.377×10^{-3}	0.181×10^{-2}
10 days				
Dose	0.122×10^{-3}	0.165×10^{-2}	0.313×10^{-1}	0.1501
50% risk	0.165×10^{-5}	0.223×10^{-4}	0.422×10^{-3}	0.202×10^{-2}
20% risk	0.663×10^{-6}	0.895×10^{-5}	0.169×10^{-3}	0.810×10^{-3}
25 days				
Dose	0.749×10^{-4}	0.101×10^{-2}	0.191×10^{-1}	0.916×10^{-1}
50% risk	0.101×10^{-5}	0.136×10^{-4}	0.258×10^{-3}	0.123×10^{-2}
20% risk	0.404×10^{-6}	0.546×10^{-5}	0.103×10^{-3}	0.494×10^{-3}
50 days				
Dose	0.442×10^{-5}	0.596×10^{-4}	0.112×10^{-2}	0.540×10^{-2}
50% risk	0.596×10^{-7}	0.805×10^{-6}	0.152×10^{-4}	0.729×10^{-4}
20% risk	0.238×10^{-7}	0.322×10^{-6}	0.608×10^{-5}	0.291×10^{-4}
120 days				
Dose	0.109×10^{-5}	0.147×10^{-4}	0.773×10^{-3}	0.299×10^{-2}
50% risk	0.147×10^{-7}	0.199×10^{-6}	0.104×10^{-4}	0.404×10^{-4}
20% risk	0.59×10^{-8}	0.796×10^{-7}	0.417×10^{-5}	0.161×10^{-4}
400 days				
Dose	0.388×10^{-6}	0.524×10^{-5}	0.99×10^{-4}	0.474×10^{-3}
50% risk	0.524×10^{-8}	0.707×10^{-7}	0.133×10^{-5}	0.648×10^{-5}
20% risk	0.209×10^{-8}	0.283×10^{-7}	0.534×10^{-6}	0.256×10^{-5}

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